FY 2019 Consolidated Innovative Nuclear Research FOA Workscopes

APPENDICES

Appendix A: Workscopes for U.S. University-led Program and/or Mission Supporting R&D Projects

Appendix B: Workscopes for U.S. University-, National Laboratory-, or Industry-led Program and/or Mission Supporting R&D Projects

Appendix C: Workscopes for U.S. University-led Integrated Research Project (IRP) R&D

Appendix A: Workscopes for U.S. University-led Program and/or Mission Supporting R&D Projects

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

INNOVATIVE NEW NICKEL ALLOYS FOR MOLTEN SALT REACTOR STRUCTURAL APPLICATIONS (RC-1)
(FEDERAL POC – SUE LESICA & TECHNICAL POC – SAM SHAM)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The Molten Salt Reactor Experiment (MSRE) was operated at Oak Ridge National Laboratory during the 1960s for about five years. It deployed a uranium fuel dissolved in a fluoride-salt medium, with graphite as a moderator. A nickel alloy with about 6%Cr and molybdenum as a solid solution strengthening agent was developed and deployed for all structural applications in the MSRE. The alloy was later commercialized as Hastelloy N. During these experiments and in post-decommissioning characterization of material behavior, it was determined that the most significant challenges for structural materials are embrittlement, from helium introduced by transmutation of Ni, and corrosion and grain boundary embrittlement from the fission product tellurium. The MSRE had an outlet temperature of approximately 650°C. The mechanical properties of Hastelloy N are not sufficient to support long-term operation of an MSR above this temperature.

An NEUP award on developing the next generation MSR structural materials based on the high entropy alloy concept was made in FY18. The objective of the FY19 project is to identify existing or develop new nickel alloy(s) that can be used for welded construction of structural components for solid and liquid fueled MSRs. Characteristics of the new nickel alloy(s) to be considered include, but not limited to, high temperature strength, fuel salt compatibility, irradiation damage resistance (including helium generation from n,α reactions with thermal neutrons), fission products embrittlement, and weldability, all for the desired life times of the components. For structural applications with long service lives, microstructural stability of the candidate nickel alloys in the MSR environment is an important attribute. While not specifically a part of the scope of this Call, the long-term goal of alloys developed under this effort would be their qualification for nuclear service under ASME Section III, Division 5, hence the long-term stability, fabricability, and potential capability for commercialization of any alloys developed are important.

Innovative concepts such as exploiting nano-scale interfaces within the alloy to trap defects and helium, and novel application of high-valued experiments with integrated computation materials engineering are highly encouraged. The outcome of the project is to demonstrate the potential of the developed nickel alloy(s) to meet the challenges under the extreme environments for solid and liquid fueled MSRs and a plan for fabrication scale up and intermediate term testing to further demonstrate the capability of the developed alloy(s) to meet these challenges. While not required, interaction with MSR developers on their system requirements is highly encouraged.

SALT BEHAVIOR IN MOLTEN SALT REACTORS (RC-2)

RC-2.1: UNDERSTANDING, PREDICTING, AND OPTIMIZING THE PHYSICAL PROPERTIES, STRUCTURE, AND DYNAMICS OF MOLTEN SALT
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

With the ongoing development of molten salt reactors there is a significant need for understanding the thermochemical behavior of salt compositions. Thermodynamic models are needed to predict critical salt characteristics such as melting points, heat capacity, free energies for potential corrosion reactions, and solubilities for fission and corrosion products as function of temperature and composition. The atomic

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

composition and redox condition of the salt may change with of time as a result of fission product formation and irradiation effect. Proposals are requested to better understand, predict, and optimize the physical properties and thermochemical behavior of molten salts. The goal is to develop and use first-principles molecular dynamics simulations and computational electronic structure method to extend the limited experimental data sets in covering a broader range of chemical evolution and environments. Innovative approaches to (1) apply molecular dynamics simulations to predict thermophysical and transport properties; (2) build multi-component models for prediction of phase diagrams; and (3) develop advanced models to guide the experimental efforts to manipulate the molten salt thermophysical properties are especially encouraged.

RC-2.2: UNDERSTANDING THE STRUCTURE AND SPECIATION OF MOLTEN SALT AT THE ATOMIC AND MOLECULAR SCALE
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

To understand the effects of structure and dynamics of molten salts on their physical and chemical properties—such as viscosity, solubility, volatility, and thermal conductivity—it is necessary to determine the speciation of salt components as well as the local and intermediate structure at operationally relevant temperatures. Real-time spectroscopic and electrochemical methods can help monitoring key chemical species in solution allowing for optimization of reactor performance and lifetime. Proposals are requested to take advantage of recent breakthroughs in advanced characterization tools and instrumentation methods to provide information at the atomic and molecular scale. The goals are to determine the local structure and bonding of chemical species in salt solution and to develop innovative real-time analytical methods for microscopic and macroscopic property measurements to underpin and support molten salt reactor design and development. Innovative approaches to: (1) determine salt molecular structure using scattering and spectroscopic methods; (2) develop novel electrochemistry and spectroscopy methods for in-situ monitoring and predictive modeling; and (3) develop molten salts optical basicity scale to determine corrosivity and solubility of actinides are especially encouraged.

RC-3: LIQUID METAL-COOLED FAST REACTOR TECHNOLOGY DEVELOPMENT AND DEMONSTRATION TO SUPPORT DEPLOYMENT
(FEDERAL POC – TOM SOWINSKI & TECHNICAL POC – BOB HILL)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The Department of Energy, national laboratories, and U.S. nuclear industry are aggressively working to revive, revitalize, and expand U.S. nuclear energy capacity. Advanced non-light water reactors such as liquid metal-cooled fast reactor concepts offer the potential for significant improvements to safety, economics, and environmental performance to help sustain and expand the availability of nuclear power as a clean, reliable, and secure power source for our nation.

This work scope seeks proposals to develop and demonstrate innovative technology options for liquid metal (sodium or lead-cooled) fast reactors for utilization in advanced reactor concepts proposed by U.S. nuclear industry. Features that offer the potential for significant benefits in reactor capital or operating cost reductions are of particular interest.

Examples of potentially beneficial cost reduction experimental and analytical work areas include:

- Development of sensors and prognostic techniques for deployment that can monitor and quantify materials degradation in liquid metal-cooled fast reactor primary systems. Of interest are technologies that are able to detect degradation early, can survive in typical liquid metal-cooled fast reactor environments over extended periods of time, and can be embedded in/on structural materials to enable structural health monitoring (e.g., nondestructive examination techniques, remote or automated

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

inspection techniques including visualization in optically opaque coolants). Consideration should be given to deployment issues that may arise, such as powering the sensor and data exfiltration needs

- Development and application of uncertainty propagation analysis techniques to quantify impacts on key liquid metal-cooled fast reactor performance parameters (e.g., burnup) and/or safety performance. Consideration should be given to how such tools and techniques can be used to optimize and justify design and safety margins
- Development of test articles for testing in the Mechanisms Engineering Test Laboratory (METL) sodium loop facility. The test articles should consider demonstration of innovative fast reactor sub-components (sensors, seals, mechanisms, etc.) or validation of key fast reactor behaviors (e.g., thermal striping) under prototypic or near prototypic conditions
- Detailed analytical performance studies of compact heat exchanger options (e.g., microchannel configurations) for lead or sodium-cooled fast reactors. Ideally this work would be coupled with experimental validation of key performance attributes
- Development of small-scale heavy liquid metal (lead or lead-bismuth) testing capabilities and/or test articles supporting lead-cooled fast reactor technology development and demonstration

Though proposals are not limited to the example work areas above, applicants should indicate how their proposed work will support current DOE, national laboratory, and/or U.S. nuclear industry liquid metal-cooled fast reactor deployment and commercialization R&D initiatives.

HTGR TRISO FUEL PARTICLE MATERIALS (RC-4)

RC-4.1: THERMOMECHANICAL PROPERTIES OF TRISO FUEL COATING LAYERS
(FEDERAL POC – MADELINE FELTUS & TECHNICAL POC – PAUL DEMKOWICZ)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

One of the main functions of TRISO fuel performance models such as the PARticle FUel ModEl (PARFUME) is calculating the probability of coating layer failure from various causes during irradiation. This includes modeling the thermomechanical behavior of the coatings in response to high temperatures, neutron-irradiation-induced changes, interactions with fission products, stresses in the particle from accumulating fission gas inventory in the buffer layer and CO gas formation which is a specific concern in UO_2 fuel kernels. A number of particle failure mechanisms have been identified based on decades of empirical observations of irradiated TRISO fuel and associated computational modeling [1]. Fuel particles are typically operated within a defined performance envelope (e.g., burnup, fast neutron fluence, and temperature) that limits these failure mechanisms to very low, acceptable levels. Accurate TRISO particle materials property data is needed to improve performance modelling predictions. While some materials properties are easily measured on as-fabricated fuel with reasonable confidence, other property information has greater uncertainty levels. Furthermore, sensitivity studies demonstrate that some TRISO fuel properties have a much greater impact on model predictions of particle failure than others [2]. Among the most critical characteristics are irradiation-induced creep and dimensional change in pyrocarbon (PyC), and, to a lesser extent, the PyC elastic moduli [2].

Post-irradiation examination of the AGR TRISO Fuel Program's UCO TRISO fuel has demonstrated that discreet failure of the SiC layer can occur at very low fractions with the outer PyC (OPyC) layer remaining intact. Such failed SiC layer particles release cesium, but the intact OPyC layer effectively retains fission gases. The dominant mechanism of SiC-layer failure has been identified as localized fission product (primarily Pd) corrosion of the SiC layer [3]. A distinguishing feature of this SiC layer corrosion mechanism, however, is that it requires mechanical failure of the inner PyC (IPyC) layer, such that the inner SiC surface is exposed, allowing fission products to concentrate in a relatively small region. This IPyC failure mechanism is driven by a strong bond between the IPyC and the buffer layer. Densification of the buffer layer in the presence of a strong buffer-IPyC

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

bond places additional stresses on the IPyC layer that causes IPyC fracture and IPyC-SiC delamination. Modeling of this buffer IPyC failure mechanism, which is not currently included in PARFUME or other TRISO fuel performance models, would require buffer strength and the interfacial buffer-IPyC bond strength information, for which no data are currently available.

Research proposals are sought that specifically focus on obtaining these specific TRISO coating layer properties: (a) Buffer: elastic modulus, tensile strength; (b) Pyrocarbon (IPyC and OPyC): elastic modulus, tensile strength, irradiation-induced creep and dimensional change; and (c) Buffer-IPyC bond strength. Other property measurements may be proposed, but the parameter must have a high importance ranking and impact on particle failure rates, as shown either in the PARFUME fuel performance sensitivity study [2] or based on empirical observations of TRISO fuel behavior under irradiation.

Data should be obtained at VHTR reactor-relevant temperatures using the most prototypical un-irradiated and irradiated samples possible. In cases where no data currently exist (specifically, buffer strength, buffer elastic modulus, and buffer-IPyC bond strength), un-irradiated TRISO properties would be useful as a starting point. In that case, development and demonstration of methods to measure buffer strength and buffer-IPyC bond strength on un-irradiated, prototypic TRISO materials is valuable. For TRISO pyrocarbon studies, it is critical that properties data be determined as a function of neutron fluence and damage (dpa). If surrogate material specimens are produced and used for this research, then the specimen microstructure should be comparable to AGR fuels and their irradiation damage (dpa) should be comparable to AGR fuel irradiation campaign doses. The AGR program has irradiated hundreds of thousands of TRISO particles to burnups ranging from about 5% to 20 % fissions per initial metals atoms (FIMA), fast neutron fluences ranging from 1.0E25 n/m^2 to 4.5E25 n/m^2 (E > 0.18 MeV), and irradiation temperatures ranging from approximately 900 to 1250°C. Any data collected should encompass a realistic range of HTGR normal operations and accident temperatures and neutron fluence levels. All experiments and post-irradiation microscopy tasks must be performed to NQA-1 standards. Data, experiments, fuel performance computational modelling information, and any calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS).

[1] J.T. Maki et al., "The challenges associated with high burnup, high temperature and accelerated irradiation for TRISO-coated particle fuel," J. Nucl. Mater. 371 (2007) 270-280.

[2] William F. Skerjanc, Blaise P. Collin, Assessment of Material Properties for TRISO Fuel Particles used in PARFUME," INL/EXT-18-44631, Rev. 0, 2018.

[3] J.D. Hunn et al., Detection and analysis of particles with failed SiC in AGR-1 fuel compacts, Nucl. Eng. Des. 306 (2016) 36-46.

RC-4.2: EFFECT OF NEUTRON IRRADIATION ON FISSION PRODUCT TRANSPORT THROUGH TRISO PARTICLE SILICON CARBIDE COATING LAYER
(FEDERAL POC – MADELINE FELTUS & TECHNICAL POC – PAUL DEMKOWICZ)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Research on neutron irradiation damage tolerance of silicon carbide (SiC) and SiC composites, used in various nuclear applications, such as tri-structural-isotropic (TRISO) fuel and SiC cladding for Accident Tolerant Fuel for light water reactors (LWR), has not investigated the irradiation-induced polymorphism effects in SiC. Recently, the Advanced Gas Reactor TRISO Fuel microscopy researchers have discovered that intra-granular precipitation of fission products in the TRISO SiC coating layer occurs because of a dual step nucleation mechanism that involves the cubic (β) to hexagonal (α) SiC polymorphic transition, and subsequent transition of α-SiC into fission product precipitates [1]. This intra-precipitation behavior has not been explained completely by any known nucleation mechanism, but may provide unique beneficial metallurgical traits for mitigating the effects of fission product precipitate damage (i.e., palladium that attacks SiC) in TRISO fuel during normal operations and higher temperature accident conditions, up to 1600 °C.

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

Research proposals are sought that focus on neutron irradiation-induced polymorphism in the TRISO SiC coating layer using experimental methods and advanced microscopy (e.g. SEM, TEM) that compare un-irradiated and irradiated TRISO particles with different damage (dpa) levels, using the most prototypic SiC materials available from US Advanced Gas Reactor program sample archives. While the emphasis is on carefully designed experiments, comparison of results with computational models of SiC polymorphic transition effects on fission product transport as a function of neutron damage behavior is beneficial. Any data collected or equations developed should permit application to a realistic range of HTGR operations and accident temperatures and neutron fluence levels. All experiments, microscopy, and modeling tasks must be performed to NQA-1 standards. Data, experiments, computer modelling information, and calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS).

[1] S. Meher, I. J. van Rooyen, and T. M. Lillo: A novel dual-step nucleation pathway in crystalline solids under neutron irradiation, Scientific Reports, 8 (2018) 98. Available at: https://www.nature.com/articles/s41598-017-18548-8.

RC-5: EXPERIMENTAL VALIDATION OF HIGH TEMPERATURE GAS REACTOR (HTGR) SIMULATIONS

(FEDERAL POC – DIANA LI & TECHNICAL POC – HANS GOUGAR)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Experimental validation of High Temperature Gas Reactor (HTGR) simulations is focused on providing data of high temperature gas-cooled reactor (prismatic or pebble bed) phenomena for the validation of system and computational fluid dynamics models. These phenomena are relevant to core safety and performance.

One of the scenarios of interest to core designers and safety analysts is a break in the primary coolant boundary that leads to depressurization of the primary loop. If no mitigating action is taken, any oxygen remaining in the reactor cavity and surrounding building can enter the primary through the break and damage the graphite structures via oxidation. (The flow path within the primary loop also has a strong influence on the amount of air actually reaching the graphite core structures but this is particularly design dependent and beyond the scope of this call.) A full-blown 'double-ended guillotine break' of the primary coaxial coolant duct is a beyond design basis event and is not within the scope of this call.

The amount of cavity air entering the pressure vessel is a complex function of the primary helium inventory (which displaces, and mixes, with the air in the cavity), the break location, size, and orientation, and the venting pathways in the reactor building itself as these determine the fraction of air forced from the building during the blowdown and the rate at which fresh air would be subsequently replenished in the reactor cavity over long periods of time [1] as the reactor gradually cools. An experiment sponsored by the Department of Energy and the NGNP Alliance yielded preliminary results and insights but a full parametric study has yet to be performed. Of particular interest is the spatial distribution of air and helium in each reactor building cavity during the blowdown. The overall effects of reactor building vent path options should also be considered (i.e., venting from the top or bottom of the building cavity(ies)).

Parametric studies should investigate the concentration of oxygen in the vessel to various small and medium-sized breaks in the primary system (pressure vessel, crossduct, or steam generator vessel), the orientation of those breaks, and alternate ventilation pathways. Principal Investigators are encouraged to consider the recommendations of the NGNP Alliance contained in reference [2].

The General Atomics 350 MWt MHTGR [3] should be used as the basis for scaling the experimental facility.

Principal Investigators are encouraged to consult with US-based HTGR vendors (Framatome, X-Energy, USNC)

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

to refine the experiment design and test matrix.

All experiments must be performed to NQA-1 standards. Data, experiments, and calculations shall be submitted to the Idaho National Laboratory's NGNP Data Management and Analysis System (NDMAS). Assistance shall be provided by Idaho National Laboratory for NDMAS use and ensuring NQA-1 standards are properly established.

[1] Yang, Se Ro & Kappes, Ethan & Nguyen, Thien & Vaghetto, Rodolfo & Hassan, Y. (2018). Experimental study on 1/28 scaled NGNP HTGR reactor building test facility response to depressurization event. Annals of Nuclear Energy. 114. 154-164. 10.1016/j.anucene.2017.12.023.

[2] NGNP Deliverable to DOE: Project Recommendations – Evaluation and Testing of HTGR Reactor

Building Response to Depressurization Accidents, AREVA Document 12-9273530-00, July 2017.

[3] Preliminary Safety Information Document for the Standard MHTGR. Volume 1, United States: N. p., 1986. Web. doi:10.2172/712676.

RC-6: FLUORIDE SALT-COOLED HIGH TEMPERATURE REACTOR (FHR)

The Fluoride Salt-cooled High Temperature Reactor (FHR) focus area is seeking to address either one of the two areas discussed below. All work performed for this call must be performed to NQA-1 standards.

RC-6.1: USED FHR PEBBLE FUEL STORAGE AND HANDLING
(FEDERAL POC – DIANA LI & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Used FHR fuel pebbles will require cooling after removal from the core prior to being able to rely on natural circulation air cooling in dry casks. Also, the pebbles will need to be transferred from the outlet of the fuel handling system to the cooling/storage system, which may be in a separate low-leakage containment environment. FHR fuel pebbles will be both smaller and have higher heat loads than those of High Temperature Gas Reactors due to their higher overall fuel loadings. This call seeks evaluation of technologies that support safe and effective used fuel pebble handling and storage prior to dry storage. For example, light water reactors transfer used fuel assemblies to a spent fuel pool for storage until they are cool enough for dry cask storage.

For the FHR, development and validation of models and simulations of passively-safe, used FHR pebble storage and handling technology is requested. The simulations should include fissile materials accountancy including developing performance uncertainty estimates (see INL/EXT-12-26561 –"Safeguards-by-Design: Guidance for High Temperature Gas Reactors (HTGRs) With Pebble Fuel"). Provide enough detail to determine necessary storage times to meet requirements for dry storage, storage configuration of pebbles, and ensure there is sufficient capacity to store pebbles for the entire lifetime of the FHR. Evaluations should also identify any necessary modifications to existing used fuel storage/transport casks for storage of FHR pebble bed fuel. Additionally, this call seeks techniques for handling and safely storing cracked/broken pebbles.

RC-6.2: EXPERIMENTAL VALIDATION OF PASSIVE DECAY HEAT REMOVAL TECHNOLOGY
(FEDERAL POC – DIANA LI & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

A key element of FHR passive safety is the ability to reject decay heat to the environment indefinitely following a station blackout event. While both Direct Reactor Auxiliary Cooling System (DRACS) AND Reactor Vessel Auxiliary Cooling System (RVACS) type decay heat removal mechanisms are possible for FHRs, FHR RVACS systems would be similar to those of other high-temperature reactors, which have been recently modeled at

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

ANL's natural convection shutdown heat removal test facility
- https://www.ne.anl.gov/capabilities/rsta/nstf/index.shtml.

Molten salt DRACS systems, however, have not previously been experimentally demonstrated. Demonstration of a reduced capacity, DRACS-type natural circulation decay heat rejection system employing prototypic materials and temperatures is requested. The system should include the ability to simulate cold start-up, normal reactor operations, transition to accident conditions, and extended duration operation and confirm the system will sufficiently remove heat without salt freezing. The system should also include tritium release mitigation mechanisms. System design, fabrication, and operation should be adequately documented to provide information that could be used in a 10CFR50 Appendix B program.

RC-7: MOLTEN SALT REACTOR TECHNOLOGIES

RC-7.1: FUEL SALT SAMPLING TECHNOLOGY DEVELOPMENT
(FEDERAL POC – BRIAN ROBINSON & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

A key technology for MSR operations is the ability to sample the hot, highly-radioactive fuel salt and to introduce additional (e.g. fuel or redox control) materials. Key parameters such as the progress of corrosion, fissile material consumption and isotope distribution, fuel salt redox condition, and in-leakage of coolant salt can be assessed by measuring fuel salt composition. The Molten Salt Reactor Experiment (MSRE) employed a sampler-enricher to enable fuel salt access. While the MSRE's sampler-enricher was generally functional, it had less than desirable reliability. [1] Proposals are requested to develop and demonstrate, in a non-radioactive environment, a modern equivalent to MSRE's sampler-enricher with improved reliability and potential to serve as a technology model to guide deployment in future MSRs.

[1] R. B. Gallaher, Operation of the Sampler-Enricher in the Molten Salt Reactor Experiment, ORNL-TM-3524, October 1971, https://www.osti.gov/biblio/4731130

RC-7.2: EVALUATION OF 316SS LIFETIME IN MSRS
(FEDERAL POC – BRIAN ROBINSON & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

MSR designers are considering employing mature iron-nickel-chrome alloys such as stainless steel 316 along with active salt chemistry control as their salt wetted structural material. Only limited amounts of corrosion are anticipated with 316SS given the low solubility of the alloy constituent elements (non-oxidative corrosion) provided the salt is maintained in a reducing condition. Nevertheless, the high salt temperature, neutron damage, and mechanical service requirements will cause the material properties to degrade over time. Moreover, both generalized and grain boundary corrosion are expected to weaken the surface making it more vulnerable to erosion. One of the key design properties for any molten salt heat transport system is the maximum allowable fluid velocity. Understanding material aging under service conditions will support establishing an evidence-based flow specification. Experimental projects are sought to evaluate the combined corrosion and mechanical stress impact on SS316 component service lifetimes and design limits.

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

RC-7.3: RADIATION HARDENED VISION SYSTEMS
(FEDERAL POC – BRIAN ROBINSON & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

MSRs will need to perform remotely guided, highly-automated, operations and maintenance (O&M) with remote tooling. Maintenance was called out as an especially difficult problem for MSRs in the AEC's review of the MSBR in 1972. [1] Optically based systems remain the most important sensing modality for guiding remote O&M. A key element of enabling the automation system or the operator to perform tasks is to provide real-time 3D visual updates of the positioning of the tooling, components, and surround structures. Depending on the local shielding employed, the MSR containment environment may have very high radiation dose rates. Radiation hardened remote tooling and operations have been developed in support of O&M in multiple prior high-radiation environments. Relatively recently, the Spallation Neutron Source target handling facilities and ITER have developed radiation hardened remote O&M technology. Demonstration of a multi-camera, radiation hardened 3D vision system to continuously update the in-containment model status is requested. Demonstration of techniques to repair and/or replace vision system components within containment is also requested.

[1] U.S. Atomic Energy Commission Division of Reactor Development and Technology, An Evaluation of the Molten Salt Breeder Reactor, WASH-1222, September 1972

RC-7.4: MOLTEN SALT MECHANICAL FILTERS
(FEDERAL POC – BRIAN ROBINSON & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Fission products will be generated in fuel salt. Not all of the fission products will be soluble. Those that are not will be suspended in the salt and tend to plate out onto surfaces within the system. While plating out corrosion resistant materials onto surfaces would, to a limited extent, be considered a positive/protective effect, it is anticipated that reactor vendors will mechanically filter fission products out of the salt. Additionally, under certain conditions, fissile materials may also plate out onto filters. Sintered nickel is the leading candidate structure to serve as a mechanical filter. Filtering out radionuclides, however, has a number of complex interrelated issues such as monitoring filter condition and performance, introducing and removing the highly-radioactive filter (if not a lifetime component), cooling and shielding the filter once removed, and surveying the filter for fissile material control and accountability. Experimental projects are sought that demonstrate fuel salt mechanical filter performance and operational issues using non or low radioactivity materials.

RC-7.5: SHUT-OFF VALVE TECHNOLOGY DEVELOPMENT
(FEDERAL POC – BRIAN ROBINSON & TECHNICAL POC – DAVID HOLCOMB)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

MSRs rely on multiple barrier layers to provide defense-in-depth against radionuclide release. The primary coolant salt will be operated at a somewhat higher pressure than the fuel salt to cause in-leakage in the event of heat exchanger tube failure. Nevertheless, the primary coolant salt lines penetrate radionuclide containment layers providing a potential barrier bypass route. The ability to provide high reliability closure to the primary coolant salt lines on-demand, thus, decreases the risk of radionuclide release. These valves may be a safety-related item as they could be relied upon to mitigate the impact of postulated accidents. Consequently, it would be desirable for them to remain operable even under beyond design basis event conditions. High-reliability, molten salt, safety-related shutoff valves with local activation energy storage have not previously been developed or demonstrated. It is requested to design and demonstrate (on a molten salt flow loop) MSR coolant salt shutoff valves whose component technologies would be suitable for qualification under a 10CFR50 Appendix B quality assurance program.

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

RC-8: PLANT MODERNIZATION R&D PATHWAY

RC-8.1: DIGITAL INSTRUMENTATION AND CONTROL QUALIFICATION
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Digital I&C qualification continues to be a major impediment to addressing the reliability and obsolescence issues in legacy analog I&C systems for the operating LWR fleet. In particular, the issue of digital common cause failure (CCF) has been difficult to address and has been the reason some nuclear plant operators have deferred upgrades of these critical plant systems, opting rather to maintain them with costly engineering and maintenance efforts.

To enable the LWR fleet a path forward with digital upgrades, a qualification method to address CCF needs to be available. There are two proposed qualification methods that require further investigation. They are 1) Testability and 2) Elimination of CCF triggers. The focus of this research is to evaluate the Elimination of CCF triggers. This analysis should focus on determination of an approach that will ensure latent digital defects are not concurrently triggered in multiple digital functions that are assumed to be independent. The outcomes of research would be to develop qualification methods and processes that would potentially address the technical and regulatory aspects of digital qualification, with regard to digital CCF.

RC-8.2: ANALYTICS TO SUPPORT EQUIPMENT CONDITION MONITORING
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Equipment monitoring is one of the main contributors to maintenance costs in nuclear power plants. Whether for diagnostics or prognostics, condition monitoring is mostly based on manual efforts to collect and analyze measurements using conventional sensors and methods. Decision making is driven by a combination of periodic inspections based on vendor specifications, industry standards, and human experience.

A need to automate the phases of data collection and analytics-based decision making has emerged to reduce the workforce cost, optimize the maintenance activities during normal operation and outage conditions, maximize the maintenance activities value, and reduce risk and human errors.
In line with the vision to automate the condition monitoring process, research will focus on identifying novel sensors, automation technologies, and data analysis methodologies for optimized condition monitoring in nuclear power plants. The results of this research will help the LWR fleet drive down the costs of maintenance by providing a scalable and adaptive data infrastructure solution needed to enable implementation of a risk informed maintenance program.

RC-9: RISK-INFORMED SYSTEMS ANALYSIS R&D PATHWAY

RC-9.1: USE OF INTEGRATED PRA AND MECHANISTIC TOOLS TO ACCELERATE DEPLOYMENT OF ADVANCED TECHNOLOGIES TO LWRS
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – RONALDO SZILARD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Transformative nuclear technologies are at various stages of development aiming at improving the economic performance, while maintaining and enhancing the superior safety performance, of the existing fleet of nuclear power plants. These new nuclear technologies include, but are not limited to: accident tolerant fuel, increased enrichment and fuel discharge burnup, fuel cycle length extension, flexible operations, passive cooling, digital instrumentation and control upgrades, and for applications other than electricity generation. Advanced modeling and simulation tools can play an important role to accelerate the deployment of these technologies. The technological and economic risks associated with the deployment of these technologies need to be thoroughly evaluated in design to ensure they meet the safety goals of plants and achieve economic benefits needed before they are deployed. Risk assessments of nuclear power plants are typically performed by combining probabilistic risk assessment (PRA) methods with mechanistic analysis methods. PRA methods not only estimate risk metrics such as core damage frequency (CDF) or large early release frequency (LERF) but also determine what the most probable accident sequences are and the components that contribute the most to the overall plant risk. Mechanistic methods employ multi-physics simulation tools in order to assure that plant safety systems can prevent core damage condition for a given set of accident conditions.

The research of this call will be to develop and apply an integrated PRA and mechanistic simulation framework to reduce the time, cost and uncertainty associated with developing, demonstrating, and deploying new nuclear technologies including increased enrichment and fuel discharge burnup, flexible operations and extending the operations of existing nuclear power plants in non-electricity applications and markets. The resulting integrated evaluation framework should enable plant system configuration variations to be studied with speed and precision, including detailed risk and benefit assessments of introducing these advanced nuclear technologies into current LWR plants to achieve both safety and operational performance enhancements. Proposals that combine the analysis methodology development and experimentation into a unified approach to address key technical and economic challenges are strongly encouraged.

RC-9.2: Risk-Informed Asset Life Cycle Management And Maintenance Optimization
(Federal POC – Alison Hahn & Technical POC – Ronaldo Szilard)
(Eligible to Lead: Universities Only)
(Up to 3 years and $800,000)

Plant asset management consists of methods designed to reduce costs and manage plant financial risk. Such methods include a large variety of models and data including:

- System Structure and Components (SSCs) aging
- Test and maintenance data
- Long term capital asset management

The call is seeking research proposals designed to reduce nuclear power plant operational costs by identifying plant asset management data that is needed to develop tools for risk-informed insights that can improve plant asset management. Methods and tools will be developed to allow risk-informed integrated business case assessments for SSCs that can more effectively and efficiently address equipment performance. This may include the use of improved data analytics with a focus on their use and management of data that are currently available.

Proposals should address the development of advanced predictive and data analysis models. This research should integrate with the RAVEN statistical framework. Proposals should address the integration of results in a coherent plant asset management framework.

The proposals may also consider how to use the results to assist in ranking major capital refurbishment and replacement of SSCs for the lifetime of a plant, to risk-inform the capital improvements that includes assessment of expected useful life. The methods and tools will be used to develop an optimized schedule for capital replacement or refurbishment of major equipment for extended plant operations.

RC-10: Materials Research Pathway

RC-10.1: Rapid, multi-modal characterization of concrete's degradation

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES

TOLERANCE IN SUPPORT OF SECOND LICENSE RENEWALS OF NUCLEAR POWER PLANTS
(FEDERAL POC – ALISON HAHN & TECHNICAL POC – KEITH LEONARD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Structural concrete components in nuclear power plants may face significant structural integrity challenges due to exposure to neutron and gamma irradiation, super-ambient temperatures, and moisture ingress. The concrete's tolerance to degradation in such environments is often dictated by the physical- and chemical-degradation of its aggregate constituents; i.e., as a function of their mineralogy, cement chemistry, etc. Existing protocols for degradation assessment, e.g., based on ASTM standards, are unsuited for assessing concrete's tolerance to degradation in such extreme environments.

Therefore, it is proposed to develop new science-based, rapid characterization methods that exploit multi-modal imaging, compositional analysis, and related methods to assess the tolerance of structural concrete and its constituents to degradation, i.e., especially in the context of second license renewal applications. Degradation modes of relevance include, but are not limited to irradiation damage resistance and chemical inertness at super-ambient temperatures in the presence of moisture. The systematic integration of pioneering experimental analyses with the integrated computational materials engineering software MOSAIC (Microstructure-Oriented Scientific Analysis of Irradiated Concrete) currently under development by the DOE Light Water Reactor Sustainability Program, for characterization method development and material analysis is highly encouraged.

RC-10.2: MODELING OF HELIUM BUBBLE DEVELOPMENT DURING WELDING OF IRRADIATED METAL ALLOYS

(FEDERAL POC – ALISON HAHN & TECHNICAL POC – KEITH LEONARD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Welding is commonly used during repair and upgrades of nuclear components. However, as the service lives of nuclear reactors are extended, the amount of transmuted helium in reactor structural materials increases, eventually reaching levels at which conventional welding technologies cannot be used reliably. Helium induced cracking in the heat affected zone has been observed at helium concentration levels below 10 atomic parts per million, a level that can be reached after approximately 40 - 60 years of operation depending on material and location. While friction stir welding and an advanced variation on laser welding are being evaluated for the repair of irradiated materials in the DOE Light Water Reactor Sustainability Program, a fundamental understanding of the effect of weld heat input on the diffusion and coalescence of helium on grain boundaries in the heat affected zone for a given material and weld conditions are not well known. Proposals are sought for a comprehensive model of helium bubble growth on grain boundaries for a given material condition (type, microstructure, reactor aging condition) and heat input (welding technique), that accounts for transient high temperature stresses generated during welding and grain boundary cohesivity, to provide industry with a clear evaluation of what weld techniques and parameters should be considered in the repair of reactor components. Validation of model under combined temperature and stress conditions are essential in the research effort. Materials of interest include type 304L and 316L stainless steel as well as alloy 182 weld metal.

RC-11: SPECIAL PURPOSE REACTORS R&D
(FEDERAL POC – REBECCA ONUSCHAK & TECHNICAL POC – SHANNON BRAGG-SITTON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

The Advanced Reactor Technologies Program supports technology development efforts for MW-class, very small modular reactors (e.g., <20 MWt). These “special purpose reactors” are of interest for potential national

PROGRAM SUPPORTING: NUCLEAR REACTOR TECHNOLOGIES
security and deployable (terrestrial) power applications for defense, commercial, or industrial use. Applications are sought for technologies that support portable compact reactors that would be used in a microgrid configuration. Topics of interest only include technologies with features specific to special purpose reactors. Some specific areas of interest include use of advanced manufacturing to support factory manufacture of reactor and system components, reducing the time and cost associated with fabrication and assembly of the energy system; advanced instrumentation and control approaches that support semi-autonomous or autonomous control; and novel power conversion systems, static or dynamic, that improve on the current state of the art, as well as the associated heat exchanger designs. Specific technology proposals could be relevant to conversion of heat from a fission heat source to electrical power and/or direct use of heat for other applications (e.g. district heating, water purification, hydrogen generation, synfuels production, etc.).

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

MATERIAL RECOVERY AND WASTE FORM DEVELOPMENT (FC-1)

This program element develops innovative methods to separate reusable fractions of used nuclear fuel (UNF) and manage the resulting wastes. These technologies, when combined with advanced fuels and reactors, form the basis of advanced fuel cycles for sustainable and potentially growing nuclear power in the U.S.

FC-1.1: ELECTROCHEMICAL SEPARATIONS
(FEDERAL POC – STEPHEN KUNG & TECHNICAL POC – MARK WILLIAMSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The solubility limit of fission products in molten chloride salts becomes one limit that impacts the timing for salt treatment and recycle, or disposal. Solubility data as a function of temperature and composition is available for many binary and ternary chloride systems especially those that contain alkali and alkaline earth chlorides. However, similar data is lacking for complex multicomponent chloride salt systems especially those involving the lanthanide and actinide elements. We are seeking proposals to establish fundamental thermochemical data in chloride salt systems with an emphasis on multicomponent solubility of the fission product chlorides. Experimental work may be supplemented by thermodynamic data simulation to yield a more complete understanding and predictive models of the solid-liquid equilibria in the complex chloride systems.

FC-1.2: MATERIALS RECOVERY
(FEDERAL POC – DAN VEGA & TECHNICAL POC – TERRY TODD)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Solvent Extraction Chemistry and Radiation Chemistry
Aqueous recycling of used nuclear fuel using the PUREX process is being used industrially for several decades. However fundamental knowledge and understanding of the chemical speciation and partitioning of multivalent cations (e.g. Np, Tc, Zr) in advanced PUREX based extraction process using n-Tri-Butyl-Phosphate (TBP) as solvent under high gamma irradiation fields is required to improve processes being currently studied such as CoDCon (ConDeContamination process). Gamma radiation is known to produce radicals that can affect the oxidation states of multivalent cations in solution. This research effort will provide insight into the speciation of multivalent cations under gamma irradiation dose as well as the effects of the speciation on partitioning in a tri-butyl phosphate separations process (e.g. PUREX, COEX, CoDCon, etc).

FC-1.3: WASTE FORMS DEVELOPMENT AND OFF-GAS CAPTURE
(FEDERAL POC – KIMBERLY GRAY & TECHNICAL POC – JOHN VIENNA)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

FC-1.3a: ADVANCED WASTE FORM DESIGN, PROCESSING, AND TESTING - High-level waste (HLW) from separated commercial nuclear fuels will need to be processed into durable waste forms with predictable long-term performance. The international reference treatment technology is vitrification into borosilicate glass that is known to have many advantages including ease of processing and tolerance to variation in waste compositions. Modern nuclear fuel recycling processes generate HLW with narrow variation in composition compared to past U.S. practices with defense reprocessing. As a result, ceramic waste forms can be an effective means of immobilizing newly generated HLW. In addition, modern materials fabrication techniques may enable low cost processes to fabricate HLW forms applicable to high radiation environments (e.g., low handling, easy containment, minimal access to fine particles).

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

The focus of this research is to develop a robust waste form for commercial fuel recycling HLW stream and demonstrate low-cost processing methods that are applicable to high radiation environments. The successful waste form would allow for higher waste loadings than typical HLW glass compositions and be processable in plants with lower construction (e.g., size) and operating (e.g., manpower) costs

FC-1.3B: IODINE CAPTURE FROM HIGHLY OXIDIZING OFF-GAS STREAMS – US regulations could require the removal of both iodine and tritium from the off-gas stream of a used nuclear fuel (UNF) reprocessing facility. Advanced tritium pretreatment is a pretreatment step that uses high concentrations of NO_2 (60% to 80%) in a gas stream to volatilize tritium and iodine from the oxidized UNF prior to traditional dissolution. The gaseous effluent from this process would then require treatment to remove tritium and iodine, but high levels of NO_2 could have a detrimental effect on the ability of various solid sorbents to remove the volatile radionuclides. For iodine, the sorbents considered have been reduced silver mordenite (Ag^0Z), silver-functionalized silica-aerogel (AgAerogel), and silver-nitrate-impregnated alumina (AgA). Prior research has demonstrated that exposure to high concentrations of NO_2 can reduce the iodine loading capacity of Ag^0Z by > 90% when exposed for 1 week. Research in Japan has demonstrated that AgA is more robust to NO_2 exposure than AgZ, however, iodine capture tests in a recirculating high NO_2 system indicated only limited recovery with the iodine depositing elsewhere in the system. The focus of this research effort is the development of a robust iodine sorbent for use in this highly oxidizing environment. Characteristics of a suitable sorbent material include rapid adsorption kinetics, high iodine loading (> 5-10% by weight), low cost, very high iodine retention once loaded, high radiation stability, and producible in a mechanically-robust, engineered form. The proposed iodine loaded sorbent material must also have a direct pathway to a suitable waste form for ultimate disposal in a deep geologic repository.

ADVANCED FUELS (FC-2)

FC-2.1: POST IRRADIATION EXAMINATION (PIE)/NON-DESTRUCTIVE EXAMINATION (NDE) TECHNIQUES FOR CORROSION THICKNESS MEASUREMENTS ON ATF CLADDINGS (COATED-ZR, FECRAL, SIC)

(FEDERAL POC – FRANK GOLDNER & TECHNICAL POC – JASON HARP)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Proposals are solicited for techniques or a suite of techniques that can be applied to evaluate the surface oxide, surface layer adhesion and sub-surface features, such as micro-cracks and delamination, for a wide variety of accident tolerant cladding concepts. Techniques developed should be demonstrated to work in a remote, high radiation environment. Signals from the demonstrated techniques should travel greater than 30 feet from sensor to data collection equipment and must pass through a hermetically sealed feedthrough.

Coating adhesion and oxide layer thickness are both important parameters that need to be evaluated for new candidate cladding materials for light water reactors. Coated zirconium cladding is a leading candidate for near term deployment in light water reactors. The coating layer provides coping time benefits in the event of a severe accident. The in-pile corrosion and adhesion properties of coated Zr are not currently known. After irradiation the extent of corrosion and adhesion can be evaluated destructively with optical microscopy, but the sampling frequency of this technique is quite small. Another near-term cladding concept is FeCrAl alloys. This material generates different oxide layers depending on the chemistry of the oxidizing conditions. Composites of SiC are also being investigated and these composites generate very thin oxide layers in pressurized water conditions. The SiC-based cladding concepts also can contain various sub-surface interfaces and the integrity and damage accumulation at those interfaces will be a performance aspect that must be examined and quantified.

The oxide layer on zirconium alloys has typically been evaluated with Eddy Current. However, this technique

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

typically requires a new set of sensors for every new substrate, and the response of eddy current sensors to ferritic substrates is quite different from non-ferritic substrates. This technique is also only applicable to conductive substrates. Presently, eddy current also cannot distinguish between voids and oxide growth.

New or next-generation non-destructive techniques to evaluate both the thickness of oxide formed on the surface and the internal structure for all of these cladding concepts during steady state reactor operations are sought. Additionally, techniques for the evaluation of the adhesion of these coatings and/or oxides and/or sub-surface damage in the event of accident operations are also solicited.

FC-2.2: Studies on Accident Tolerant Control Rods and Core Components
(FEDERAL POC – FRANK GOLDNER & TECHNICAL POC –MIKE TODOSOW)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

In response to the accident at the Fukushima Daichi nuclear power plant, a significant effort was initiated by the Department of Energy, Office of Nuclear Energy (DOE-NE) in the Advanced Fuels Campaign (AFC) to develop fuels with enhanced accident tolerance, aka "Accident Tolerant Fuels (ATFs)," defined as:

> *Fuels with enhanced accident tolerance are those that, in comparison with the standard UO_2 – Zr system, can tolerate loss of active cooling in the core for a considerably longer time period (depending on the LWR system and accident scenario) while maintaining or improving the fuel performance during normal operations.*

Much of the focus to-date has been on fuels and cladding to address these objectives. However, it is recognized that retention of "functionality" of other core components, including control rods and components that affect the structural integrity/geometry of the core are also critical to successfully surviving/limiting the consequences of Beyond Design Basis Accidents (BDBAs), as well as normal operation, Anticipated Operational Occurrences (AOOs) and Design Basis Accidents (DBAs)

Proposals are sought for the neutronic, and thermo-mechanical impacts of potential future advanced materials for BWR and PWR control rods that retain/enhance the poisoning effects/requirements (worth – individual and bank, etc.) while maintaining structural integrity/functionality (e.g, ability to insert/ withdraw) during normal operation and accident conditions (temperatures, cooling, etc.). Also, materials and associated studies sought are those that address retention of the structural integrity/geometry of the core which depends on the ability of components such as grid spacers, core support plate, etc. to retain "functionality" during normal operation and accident conditions. Consultation with existing DOE-NE ATF related programs in this area is encouraged/desirable.

FC-2.3: INVESTIGATIONS INTO NON-TRADITIONAL SOLID FUELS FOR ADVANCED NON-LIGHT WATER REACTORS
(FEDERAL POC – JANELLE EDDINS & TECHNICAL POC – ANDY NELSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

A wide range of solid fuel for advanced non-Light Water reactors are presently under investigation for numerous applications. Uranium-zirconium (U-Zr) metallic fuels and mixed oxides (MOX) have been the reference fuels for the majority of historic advanced reactor concepts. Both U-Zr and MOX fuel forms have myriad benefits, but in addition both possess limitations as nuclear fuels. Alternate advanced reactor fuels have been proposed and assessed to varying degrees over previous decades, but none have been advanced to the degree of either U-Zr or MOX. Examples of alternative advanced reactor fuels include U-Mo alloys, uranium monocarbide (UC), and uranium mononitride (UN). In addition, encapsulated fuel including TRISO particles has been proposed for

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

various novel reactor designs, fast gas-cooled reactors and non-traditional applications. Each of these alternative advanced reactor fuel forms have seen limited assessment and evaluation. The limited attention paid to these and other advanced reactor fuels have resulted in slowed deployment of modern experimental and simulation tools to better understand their capabilities, uncertainties, and limitations in service.

The present call intends to stimulate broad responses that address aspects of lesser known and novel fuels for advanced reactors through leveraging of modern experimental methods and capabilities as well as modeling and simulation tools. Focus should be restricted to fuel for the following previously-defined advanced non-Light Water Reactor Concepts: High Temperature (or Very High Temperature) Gas-cooled Reactors, Lead-cooled Fast Reactors, Gas-cooled Fast Reactors, Sodium-cooled Fast Reactors and Fluoride Salt-cooled High Temperature Reactors. TRISO-like and/or encapsulated fuel particle research beyond traditional thermal high temperature helium gas-cooled TRISO-fueled reactor applications will be considered and could include: (a) alternative fuel kernels (UN, UC, MOX, TRU), (b) different coating and matrix materials besides SiC and graphitic materials, (c) novel TRISO fuel and its chemical/corrosion compatibility for use in non-water, non-helium coolant reactors, and (d) fast gas-cooled reactor applications. Traditional TRISO fuel (UO2, UCO) for thermal spectrum high temperature helium gas-cooled high-temperature reactors will not be considered.

Research objectives need to be clearly defined and appropriate for the technical readiness level of the fuel and overall technical challenges to its advancement. It is anticipated, but not required, that both experiments as well as the use of modern modeling and simulation tools will play important roles in a successful proposal. Focus may be placed on addressing either a major challenge of a single component of the fuel or extended to include the fuel/cladding/coolant system. Irradiation testing and/or post irradiation examination of fuel concepts are advantageous (if applicable to the area of study) but not required, and new methods of analyzing archival post irradiation data may constitute a component of proposed work. While the potential topic areas for the present call are broad, it will be critical that the proposed work clearly articulates the key unknowns or challenges that limit further advancement of the chosen concept and how the proposed work will surmount these challenges if successful. Any proposed research on thorium-based fuel concepts will not be considered.

FC-2.4: ADVANCED CREEP TESTING OF FERRITIC STEELS FOR REACTOR CLADDING APPLICATIONS

(FEDERAL POC – JANELLE EDDINS & TECHNICAL POC – STUART MALOY)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Long term use of fuels for advanced reactors can lead to thermal and irradiation creep of fuel cladding and other stressed structures. In addition, creep can be a contributor to dimensional changes under accident conditions in a light water reactor. Typical out of pile (thermal) creep testing requires long times (many years) and large sized specimens. Although significant testing has been performed to measure the creep of HT-9 and some oxide dispersion strengthened ferritic alloys (e.g. MA-957) for cladding applications, little data is available on newly developed ferritic alloys (ODS, FeCrAl, and tempered martensitic variants) partly because of the long time required to perform this testing.

Thus, proposals are requested to measure the thermal creep of advanced fuel cladding relevant ferritic steels using methods requiring shorter evaluation times and possibly smaller scale specimens. Proposed work should apply these improved methods to measuring thermal creep of new alloys for cladding applications such as FeCrAl alloys, ferritic ODS alloys or tempered martensitic alloys. This data is not only important experimental data but also critical to the development of multiscale models. Priority will be given to proposals that show how the data will be incorporated into relevant multiscale models. It will be advantageous (but not required) if the proposed method(s) used to measure thermal creep can also be applied to measuring irradiation creep in situ.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

FC-2.5: SEPARATE EFFECTS TESTING IN TREAT USING STANDARD TEST CAPSULES
(FEDERAL POC: KEN KELLAR & TECHNICAL POC: DAN WACHS)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $500,000)
(NSUF LETTER OF INTENT REQUIRED)

The response of materials (especially those containing fissile isotopes) often exhibit complex behavior while under irradiation. The presence of this complexity in multiple component fuel systems can make prediction of integral response of fuel designs very difficult to achieve, especially when the fuel is subjected to new environmental conditions or when the components are placed in new configurations. The unreliability of predictive calculation, and the concomitant requirement for comprehensive empirical validation testing, has been a significant barrier to technical innovation in nuclear fuel designs for decades. To break this barrier, modern nuclear fuel technology development strategies are being developed that rely on integration of micro-scale material science and thermal-mechanical engineering through advanced modeling and simulation techniques that are envisioned to accelerate the development and deployment of advanced fuel technologies. Realization of this vision requires implementing current and new nuclear science research tools in new ways to build the experimental databases that will be used to develop and qualify these tools. This includes developing the physical models input into the codes as well as the integral system data to be used in validating the result of the simulations.

Reactivation of the Transient Reactor Test (TREAT) facility creates a unique opportunity to implement this new approach. TREAT is designed to deliver a time dependent neutron flux to test specimens. The specific shape of the transient (ranging through steady-state 'flat-tops', power ramps, pulses, or combinations of each) can be selected to achieve the experimenter's desired energy deposition in the test sample. The test can be immersed in a variety of sample environments provided by specialized irradiation devices. Multi-purpose, modular test devices allow experimenters the flexibility to quickly design experiment-specific test capsules that can provide a wide array of thermal, mechanical, and/or chemical environmental boundary conditions. The response of the test sample to the nuclear stimulus while immersed in this carefully controlled environment can be readily monitored using existing qualified or user supplied instruments.

Proposals are sought that will leverage TREAT's Minimal Activation Reusable Capsule Holder (MARCH) irradiation testing system and modern modeling and simulation tools to conduct novel separate effects tests of this type. Examples may include in-situ evaluation of physical properties of fissile material while under irradiation, thermal-mechanical response of fuel system components to nuclear heating, or short-term microstructural evolution of fissile materials under irradiation. Test samples can be supplied by the experimenter or allocated from the NSUF or DOE program's library of historic materials (fresh or pre-irradiated).
This call is designated for university led investigations only. Proposals studying materials associated with high-program-interest technologies will be given the highest priority.

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Appendix D and at https://atrnsuf.inl.gov/documents/ATRNSUFStandardNon-PropUserAgreement.pdf). **The terms and conditions of the User Agreement are non-negotiable and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the FC-2.5, NEAMS-2, NSUF-1, or NSUF-2 applications.** In order to ensure compliance throughout the application review process, applicants must indicate during the pre-application and full application submission that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of an pre-application and a full application indicates the applicant will comply and agree to the terms and conditions of the User Agreement. Upon award of an NSUF supported project, the User Agreement must be signed before activities will begin on the project.

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

ADVANCED DATA INTEGRATION FOR DOMESTIC NUCLEAR SAFEGUARDS (FC-3)
(FEDERAL POC – MIKE REIM & TECHNICAL POC – MIKE BROWNE)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Molten Salt Reactor (MSR) research has continued to increase over the past 10 years. The potential safety benefits and cost savings of a comparatively simple design with respect to current commercial power reactors shows promise. However, significant research is required to establish and assess nuclear material accountancy (NMA) for liquid fueled MSRs. Specifically, online processing to remove fission products poses potential NMA challenges in a homogenous fuel/coolant system to confirm that uranium and plutonium are not diverted in the processing. Modeling efforts to quantify whether such diversions can be detected by investigating reactor response to the change of isotopic composition would benefit NMA assessments. Research proposals are sought to develop initial NMA models for simple homogenous liquid fuel MSR design with online fission product removal. University contributions are sought to:

- Utilize models to quantify observable reactor performance to isotopic composition
- Work with U.S. National Laboratory collaborators to identify potential NMA challenges

USED NUCLEAR FUEL DISPOSITION (FC-4)

FC-4.1: USED NUCLEAR FUEL DISPOSITION: DISPOSAL
(FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – PETER SWIFT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Assessments of nuclear waste disposal options start with waste package failure and waste form degradation and consequent mobilization of radionuclides, reactive transport through the near field environment (waste package and engineered barriers), and transport into and through the geosphere. Science, engineering, and technology improvements may advance our understanding of waste isolation in generic deep geologic environments and will facilitate the characterization of the natural system and the design of an effective engineered barrier system for a demonstrable safe total system performance of a disposal system. DOE is required to provide reasonable assurance that the disposal system isolates the waste over long timescales, such that engineered and natural systems work together to prevent or delay migration of waste components to the accessible environment.

Mined geologic repository projects and ongoing generic disposal system investigations generate business and R&D opportunities that focus on current technologies. DOE invites proposals involving novel material development, testing methods, and modeling concept and capability enhancements that support the program efforts to design, develop, and characterize the barrier systems and performance (i.e., to assess the safety of a nuclear waste repository). DOE will also consider proposals addressing applications of state-of-the-art uncertainty quantification and sensitivity analysis approaches to coupled-process modeling and performance assessment which contribute to a better understanding of barrier system performance and the optimization of repository performance.

Research proposals are sought to support the development of materials, modeling tools, and data relevant to permanent disposal of spent nuclear fuel and high-level radioactive waste for a variety of generic mined disposal concepts in clay/shale, salt, crystalline rock, and tuff. Key university research contributions for the disposal portion of this activity may include one or more of the following:

- Improved understanding of waste package failure modes and material degradation processes (i.e. corrosion) for heat generating waste containers/packages considering direct interactions with canister and

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

buffer materials in a chemically reducing repository environment leading to the development of improved models (including uncertainties) to represent the waste container/package long term performance.

- Improved understanding of large-scale hydrologic and radionuclide transport processes in the geosphere of relevant disposal repository environments, leading to the development of improved methodologies and models (including uncertainties) to represent these processes.

- New concepts or approaches for alleviating potential post-closure criticality concerns related to the disposal of high capacity waste packages. Development of models and experimental approaches for including burn-up credit in the assessment of the potential for criticality assessment for spent nuclear fuel permanently disposed in dual- purpose canisters that are designed and licensed for storage and transportation only.

- Development of new techniques for in-situ field characterization of hydrologic, mechanical, and chemical properties of host media and groundwater in a disposal system.
- Development of pertinent data and relevant understanding of aqueous speciation, multiphase barrier interactions, and surface sorption at elevated temperatures and geochemical conditions (e.g., high ionic strength) relevant to deep geologic disposal environments.

- Development of new and cost-effective concepts (in different geologic media -- clay/shale, salt, crystalline rock, and tuff) for sealing repository openings (e.g., shafts, tunnels, wells) to facilitate repository closure and provide required long-term waste isolation and performance.

- Identification and assessment of innovative and novel buffer materials, new methods and tools for multi-scale integration of flow and transport data, new approaches for characterization of low permeability materials, state-of-the-art tools and methods for passive characterization and monitoring of engineered/natural system component properties and failure modes and their capability to isolate and contain waste.

FC-4.2: USED NUCLEAR FUEL DISPOSITION: STORAGE & TRANSPORTATION
(FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – PETER SWIFT)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The possibility of stress corrosion cracking (SCC) in welded stainless steel dry storage canisters (DSC) for spent nuclear fuel (SNF) has been identified as a potential safety concern. The welding procedure introduces high tensile residual stress and sensitization in the heat-affected zone (HAZ), which may drive the initiation of pitting and transition to SCC growth when exposed to an aggressive chemical environment. Analysis of samples surface deposited on in-service DSCs at three near-marine ISFSIs sites have demonstrated the presence of chloride-rich salts on the outer canister surfaces (Enos et al. 2013, Bryan and Enos, 2014, EPRI, 2014, Bryan and Enos, 2015). As portions of the canister surfaces cool sufficiently, the marine atmospheric salts may deliquesce and generate an aqueous brine layer on the surface of the canisters at various locations. This aggressive environment may lead to pitting, SCC, and potentially a through-wall failure in the weldments of the canisters.

One strategy to reduce the potential for a through-wall crack is to develop repair and mitigation technologies for the identified pitting and cracks. The main incentive for development of repair and mitigation technology is to avoid the enormous cost of canister replacement, and significant safety related issues during the replacement process. The development of cost-effective repair and mitigation technologies would ensure the continuation of long-term performance of dry storage casks at ISFSIs. Development of crack repair techniques using advanced welding repair technologies in combination with advanced mitigation technologies to prevent or minimize future pitting or SCC would be essential to maintain and/or restore the mechanical integrity of the canisters under extended service conditions. Both the repair and mitigation techniques must be capable of in-service repair on loaded

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

systems, which requires low heat input, no spark source, and acceptable external forces to avoid significant reduction in mechanical strength, ignition of potential hydrogen gas inside the canister, and deformation of the canister during the repair process.

Research proposals are sought to support the understanding of the phenomena of the pitting and cracking and the efficacy of the repair and mitigation technologies for the identified pits and cracks. Key university research contributions could include one or more of the following:

- Develop numerical models and simulations for pit incubation and growth, and/or crack initiation and growth in the range of the stress, microstructure, and environmental conditions anticipated in the HAZ. Develop empirical validation techniques for the numerical models and simulations.

- Develop numerical models and simulations for possible repair and mitigation tehnologies in the range of the stress, microstructure, and environmental conditions anticipated. Develop empirical validation techniques for the numerical models and simulations.

FC-5: FABRICATION PROCESS ASSESSMENTS FOR COST ALGORITHM
(FEDERAL POC – BP SINGH & TECHNICAL POC – FRANCESCO GANDA)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

The Systems Analysis and Integration Campaign is developing algorithms for estimating the cost of nuclear energy facilities and sub-systems in a bottom-up evaluation approach. Proposals are sought for methods for assessing the complexity and cost of fabrication of large mechanical components with an "N-stamp" as opposed to standard construction, including the complexity and costs of all the steps involved in the fabrication and delivery to customers: i.e., forging, welding, machining, inspections and transportation to site, and all the necessary paperwork. The developed capability should be of direct use to the Campaign's tool-set and so there is expected to be a close interaction with national laboratory technical personnel during the project to ensure adequate integration.

FC-6: INTERFACE TOOLS FOR TRANSMUTATION DATA LIBRARY
(FEDERAL POC – BP SINGH & TECHNICAL POC – MICHAEL TODOSOW)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

In FY18 the Systems Analysis and Integration Campaign implemented an initial version of a MySQL database of fuel cycle reactor stage information. A *stage* is a complete nuclear power production subsystem, and a complete *fuel cycle system* can have more than one stage, e.g., multiple reactor types. The database consists of legacy data, including reactor stage descriptions, isotopic information, and potentially cross sections, developed by the Campaign and its predecessors, as well as data from the Nuclear Fuel Cycle Evaluation and Screening (E&S). Additionally, the database integrates other fuel cycle reactor stage transmutation information from recent analyses, such as those performed to provide information about the possible transition to a new nuclear fuel cycle.

Proposals are sought to develop portable linkage interfaces between the database and cross section generation tools, e.g. SCALE or MC2-3, and/or depletion tools, e.g. REBUS or ORIGEN, develop automated tools for verification of the data and validation of the reactor stage models, implement novel ways to utilize and/or visualize the data, generate linkages to fuel cycle simulator tools, or other novel developments. These efforts must include and follow a software quality assurance plan and must be made available to the Campaign. There is expected to be a close interaction with national laboratory technical personnel during the project to ensure adequate integration. This interaction will include providing access to the existing MySQL database to the project

PROGRAM SUPPORTING: FUEL CYCLE TECHNOLOGIES

team.

MISSION SUPPORTING: FUEL CYCLE TECHNOLOGIES

MS-FC-1: MAINTAINING AND BUILDING UPON THE HALDEN LEGACY (IN SITU DIAGNOSTICS)
(FEDERAL POC – KEN KELLAR & TECHNICAL POC – COLBY JENSEN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

With the loss of access to the Halden reactor, the nuclear research community is at risk of losing the extensive in situ diagnostic capabilities practiced at Halden. The global nuclear community is involved in using and expanding upon Halden's diagnostic achievements. The Department of Energy, Office of Nuclear Energy is interested in research and development efforts that utilize and improve upon Halden in situ diagnostics. This call is open to all viable diagnostic approaches and is not limited to the methods used at the Halden facility. Halden also demonstrated excellence in test specimen manufacture. Their work included conversion of large irradiation specimens to smaller instrumented specimens while preserving important features of the test specimen, e.g., the cracked state of irradiated fuel pellets. Another notable attribute of Halden experiments was the ability to combine multiple diagnostics on one test specimen.

Real-time in-core diagnostic instrumentation of interest include, but are not limited to: creep, crack propagation, swelling, corrosion/crud build up, temperature, pressure, flux, two-flow phase, and fission product transport. Research that enables in-core application and associated logistics is also encouraged such as focuses on miniaturization, non-contact/non-intrusive as well as innovative data transmission techniques, such as wireless methods is also encouraged.

Emphasis in awarding R&D grants will be placed on diagnostics that can most directly benefit ongoing modelling and computer simulation development and future U.S. irradiation experiments, and that measure phenomena that is difficult to assess during irradiation or post-irradiation examinations, e.g., crack propagation rates and non-linear phenomena.

PROGRAM SUPPORTING: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION (NEAMS)

SEPARATE EFFECTS IRRADIATION TESTING FOR VALIDATION OF MICROSTRUCTURAL MODELS IN MARMOT (NEAMS-2)

(FEDERAL POC: DAVE HENDERSON & TECHNICAL POC: RICH WILLIAMSON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(NSUF LETTER OF INTENT REQUIRED)
(UP TO 3 YEARS AND $500,000)

Requests are sought for innovative, separate effects irradiation tests of nuclear fuels and/or materials that would provide data important to informing and validating mechanistic, microstructure-based models of fuel behavior under development using MARMOT, the NEAMS tool for simulating microstructure evolution under irradiation. MARMOT models under active development are summarized under NEAMS 1.1 and in the MARMOT Assessment Report. Fuel systems of interest for which separate effects experiments are desired are the LWR fuel system (*i.e*., both the historic UO2 fuel and Zirconium-based cladding, as well as emerging Accident Tolerant Fuel concepts) and the SFR fuel system (*i.e*., U-Zr and U-Pu-Zr metallic fuel and steel-based cladding).

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Appendix D and at https://atrnsuf.inl.gov/documents/ATRNSUFStandardNon-PropUserAgreement.pdf). **The terms and conditions of the User Agreement are non-negotiable and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the FC-2.5, NEAMS-2, NSUF-1, or NSUF-2 applications.** In order to ensure compliance throughout the application review process, applicants must indicate during the pre-application and full application submission that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of an pre-application and a full application indicates the applicant will comply and agree to the terms and conditions of the User Agreement. Upon award of an NSUF supported project, the User Agreement must be signed before activities will begin on the project.

PROGRAM SUPPORTING: NUCLEAR ENERGY

NUCLEAR ENERGY-CYBERSECURITY RESEARCH TOPICS AND METRICS ANALYSES (NE-1)
(FEDERAL POC: TREVOR COOK & TECHNICAL POC: STEVEN HARTENSTEIN)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Cost-effectively preventing, detecting, and mitigating cyber threats to nuclear energy systems is the subject of this research. Understanding the risks associated with each design decision is fundamental to cyber protection. With the increasing application of digital instrumentation, control, and communication systems and the constant evolution of cyber security threats and technologies, there is a need for comprehensive analytical capability to model and simulate control systems and their vulnerabilities.

Proposals are sought for modeling and simulation capabilities that can inform researchers, designers and operators when assessing cyber security risks. Research of most interest will addresses characteristics and behaviors of components within embedded instrumentation and control (I&C) systems that are used within the nuclear enterprise. Another area of interest includes Integration of cyber research enabling platforms that would couple high fidelity nuclear plant simulators with emulation, hardware-in-the-loop, or human-in-the-loop instances of control and communication. Models shall capture the behavior of an I&C system, to 1) simulate characteristics of an I&C system under cyber-attack; 2) study the cyber risk impacts of upgrades and maintenance on such systems; 3) enable future nuclear energy cyber security research, and 4) facilitate nuclear facility operation education and training.

Proposals are also sought for exploring secure architectures for use in nuclear facility's digital instrumentation and control systems. Important attributes of these "secure" architectures are: eliminate or minimize common cause failure including common access attacks (access to one component of a system enables corruption of another component of the system or systems), eliminate various classes of cyber attacks (including cyber attacks that are enabled through supply chain attacks), architectures that could be resilient to a cyber attack if infiltrated, development of intrusion detection capabilities for these architectures, architectures that will allow for upgrading various components of the overall system without degrading performance or safety, architectures that are easily modeled with sufficient fidelity to support regulatory acceptance criteria, and architectures that can be manufactured in a cost effective manner.

HYBRID ENERGY SYSTEMS DESIGN AND MODELING (NE-2)

- Advanced nuclear-renewable hybrid energy systems (NHES) composed of nuclear and renewable energy sources, industrial energy users, and energy storage systems are being evaluated for their economic benefit and technical feasibility.

NE-2.1 STUDIES OF COMPONENT/SYSTEM DEGRADATION
(FEDERAL POC – MARTHA SHIELDS & TECHNICAL POC – SHANNON BRAGG-SITTON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Characterization of dynamic energy system behavior via modeling and hardware-in-the-loop (HIL) testing to capture real-time dynamic response behavior and to determine the impact of thermal cycling on components and subsystems as it relates to component and system robustness, resiliency, response rates, etc.

NE-2.2 CYBER-INFORMED SYSTEM DESIGN
(FEDERAL POC – MARTHA SHIELDS & TECHNICAL POC – SHANNON BRAGG-SITTON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Development of enhanced communications protocols that link grid operations by regional transmission organizations / independent system operators (RTO/ISO) to tightly coupled integrated systems operations to ensure that data transmission is sufficient to support optimal energy dispatch. Work should include development of front-end-controllers and associated control theory to support optimal dispatch of nuclear plant generated thermal energy to either electricity production for the grid or to support coupled non-grid processes (via thermal or behind-the-grid electrical connection). Cyber-informed engineering should be incorporated into the respective levels of data transmission and management, human monitoring and control performance, control signal processing, and device level control actions. Studies should include control hardware in-the-loop (CHIP).

NE-2.3 SYSTEMS CONTROL
(FEDERAL POC – MARTHA SHIELDS & TECHNICAL POC – SHANNON BRAGG-SITTON)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $800,000)

Develop predictive load and supply forecasting (e.g. using predictive agent-based models) to optimize energy dispatch in real time for integrated energy systems. Note that this work should involve development of system control technology/approaches.

MISSION SUPPORTING: NUCLEAR ENERGY

INTEGRAL BENCHMARK EVALUATIONS (MS-NE-1)
(FEDERAL POC: DAVE HENDERSON & TECHNICAL POC: JOHN BESS)
(UP TO 3 YEARS AND $400,000)

The International Reactor Physics Experiment Evaluation Project (IRPhEP) and International Criticality Safety Benchmark Evaluation Project (ICSBEP) are recognized world-class programs that have provided qualityassured (peer-reviewed) integral benchmark specifications for thousands of experiments. The Project produces two annually updated Organization for Economic Co-operation and Development (OECD) Nuclear Energy Agency (NEA) Handbooks that are among the most frequently quoted references in the nuclear industry. Applications are sought, within the scope of these two projects, to provide complete benchmark evaluations of existing experimental data that would be included in IRPhEP and ICSBEP handbooks, and would support current and future R&D activities.

The IRPhEP and ICSBEP Handbooks are the collaborative efforts of nearly 500 scientists from 24 countries to compile new and legacy experimental data generated worldwide. Without careful data evaluation, peer review, and formal documentation, legacy data are in jeopardy of being lost and reproducing those experiments would incur an enormous and unnecessary cost. The handbooks are used worldwide by specialists in reactor safety and design, criticality safety, nuclear data, and analytical methods development to perform necessary validations of computational models. Proposed benchmark evaluations should be of existing experimental data. Measurements of interest include critical, subcritical, buckling, spectral characteristics, reactivity effects, reactivity coefficients, kinetics, reaction-rate and power distributions, and other miscellaneous types of neutron and gamma transport measurements. A growing area of interest includes evaluation of transient benchmark experiment data for light water reactor systems, such as PWRs and BWRs.

All evaluations must be completed according to the requirements, including peer review, in the IRPhEP and the ICSBEP. DOE currently invests tens of millions of dollars each year to develop the next generation of nuclear engineering modeling & simulation tools. These tools need ad-hoc evaluated and quality-assured experimental data for validation purposes and, consequently, benchmark evaluations in support of DOE programs such as, but not limited to, TREAT, LWRS, FCT, ART, and NE's Advanced Modeling and Simulation Program (which combines application of computational capabilities from the NEAMS ToolKit and the VERA suite developed by the Energy Innovation Hub for Reactor M&S) are of particular interest to this call. To avoid duplication, please take into account ongoing work in these recent projects:

- An Integrated Research Project awarded under IRP-NE-1 in FY15 to prepare one or more TREAT transient testing benchmarks;
- Integral Benchmark Evaluation Projects awarded under MS-NE-1 in FY16 for a Molten Salt Reactor Experiment Benchmark Evaluation; and,
- Integral Benchmark Evaluation Projects awarded under MS-NE-1in FY17 for Reactor Physics Benchmark Evaluations for Power Burst Facility Experiments.
- Integral Benchmark Evaluation Projects awarded under MS-NE-1in for evaluation of TRIGA fuel characterization measurements for inclusion in the IRPhEP handbook.

NUCLEAR DATA NEEDS FOR NUCLEAR ENERGY APPLICATIONS (MS-NE-2)
(FEDERAL POC: DAVE HENDERSON & TECHNICAL POC: BRAD REARDEN)
(UP TO 3 YEARS AND $400,000)

The Evaluated Nuclear Data File (ENDF) maintained by the National Nuclear Data Program (NNDC) at Brookhaven National Laboratory (BNL) provides the most reliable and commonly used nuclear data for nuclear energy applications. However, a close and critical examination of the existing nuclear data often finds that it is inadequate for current and emerging applications.

Proposals are sought that address nuclear data needs in NE mission areas, provided that these needs are clearly demonstrated to be a limiting factor in nuclear fuel and reactor design, analysis, safety, and licensing calculations. Use of sensitivity and uncertainty analysis methods in proposed efforts is encouraged to demonstrate these needs.

Many nuclear data needs for NE may be found in the NEA Nuclear Data High Priority Request List (HPRL) (https://www.oecd-nea.org/dbdata/hprl/, which includes a broad spectrum of needs encompassing light water reactors (LWRs) as well as sodium fast reactors. Other emerging needs not yet listed on the HPRL include continued investigations of thermal scattering data in high-temperature graphite, thermal scattering data for fluorine-based molten salt reactors, and chlorine reactions for fast spectrum molten salt reactors. Additional nuclear data needs that meet documented needs for industry and DOE-NE missions are also encouraged especially as aligned with the Gateway for Accelerated Innovation in Nuclear (GAIN), Nuclear Energy Advanced Modeling and Simulation (NEAMS), Consortium for Advanced Simulation of LWRs (CASL), Advanced Reactor Technologies (ART), Fuel Cycle Research and Development (FCR&D), Transient Test Reactor (TREAT), Light Water Reactor Sustainability (LWRS) and others.

Proposals are sought that provide relevant improvements in nuclear data that address one or more stated needs by developing and demonstrating the enhancements through the entire nuclear data pipeline, from 1) new nuclear data measurements; 2) evaluation in the appropriate format (e.g. ENDF); 3) inclusion of nuclear data covariances; 4) processing into usable forms for application codes; 5) confirmation of improved predictions and uncertainties through application studies and validation; and 6) deployment through the National Nuclear Data Center at BNL for inclusion by external users in quality-assured design, analysis, safety, and licensing calculations. Partnerships with national laboratories and especially industry to clearly articulate the need for the data and to demonstrate the use of improved data in production applications are strongly encouraged.

To avoid duplication, please take into account ongoing work in these recent projects:

- Generation of thermal scattering law data for graphite and molten salts
- Generation a data format and sensitivity analysis methods for uncertainties in thermal scattering law data (see https://neup.inl.gov/SitePages/FY18_RandD_Awards.aspx)

MISSION SUPPORTING GRAND CHALLENGE (MS-NE-3)
(FEDERAL POC – TBD & TECHNICAL POC – TBD) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $400,000)

The Office of Nuclear Energy mission supports enhancing the long-term viability of the presently operating light water reactors, stimulating the development and commercialization of advanced reactor concepts, and extending nuclear energy beyond conventional electrical generation applications. Many specific challenges have been identified elsewhere in this FOA, yet many challenges remain. Applications are sought that address other issues that hinder continued operation of the existing fleet, improve the deployment potential of advanced reactor concepts or expand nuclear energy's role in meeting the nation's energy, environmental, and national security needs. Applicants must clearly outline the challenge to be addressed, the proposed solution, and the methodology that will be used to achieve the solution, including specific resources (costs and schedules) and milestones associated with the proposal's activities, as well as estimates of longer-term resources (costs and schedules) and milestones associated with implementation of the proposed solution. Proposed solutions can be at the system or component level. High-risk, high-reward ideas are encouraged.

Appendix B: Workscopes for U.S. University-, National Laboratory-, or Industry-led* Program Supporting R&D Projects

*Industry may only lead in NSUF workscopes

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

ADVANCED METHODS FOR MANUFACTURING (NEET-1)
(FEDERAL POC – TANSEL SELEKLER & TECHNICAL POC – BRUCE LANDREY)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $1,000,000)

The Advanced Methods for Manufacturing program seeks proposals for research and technology development to improve the methods by which nuclear equipment, components, and plants are manufactured, fabricated, and assembled. Proposals should support the Department of Energy's (DOE) Office of Nuclear Energy's (NE) mission to advance U.S. nuclear power in order to meet the nation's energy needs by: 1) enhancing the long-term viability and competitiveness of the existing U.S. reactor fleet; 2) developing an advanced reactor pipeline, and, 3) implementing and maintaining the national strategic fuel cycle and supply chain infrastructure.

The goal of the program is to accelerate innovations that reduce the cost and schedule of constructing new nuclear plants and make fabrication of nuclear power plant components faster, cheaper, and more reliable. The program seeks to encourage innovation that supports the "factory fabrication" and expeditious deployment of reactor technologies. Potential areas for exploration include:

1.1 FACTORY AND FIELD FABRICATION TECHNIQUES

Applications are sought for innovative technologies such as advanced (high speed, high quality) welding technologies; and modular fabrication and installation techniques.

1.2 QUALITY CONTROL TECHNIQUES AND QUALIFICATION METHODOLOGIES

Applications are sought to develop quality control techniques and qualification methodologies for advanced manufacturing processes. This should include engagement with consensus standard organizations.

The most up-to-date information on active AMM projects can be found in the 2018 NEET Advanced Methods for Manufacturing Awards Summaries on the NE website under NEET documents.

ADVANCED DIGITAL MONITORING AND CONTROL TECHNOLOGY (NEET-2)

The Advanced Sensors and Instrumentation program seeks applications for innovative technology for controls, analytics, and instrumentation of advanced reactors systems. Technology should demonstrate greater accuracy, reliability, resilience, higher resolution, and ease of replacement/upgrade capability for applications in the nuclear environment, minimizes operations and maintenance costs, and address regulatory concerns.

The proposal should indicate whether and how the proposed technology is or may be applicable to multiple reactors or fuel cycle applications, i.e. crosscutting. Proposals should support the Department of Energy's (DOE) Office of Nuclear Energy's (NE) mission to advance U.S. nuclear power in order to meet the nation's energy needs by: 1) enhancing the long-term viability and competitiveness of the existing U.S. reactor fleet; 2) developing an advanced reactor pipeline, and, 3) implementing and maintaining the national strategic fuel cycle and supply chain infrastructure.

Program Supporting: Nuclear Energy Enabling Technologies (NEET)

NEET-2.1: State of the Art I&C Technologies
(Federal POC – Suibel Schuppner & Technical POC – Craig primer)
(Eligible to Lead: University or National Laboratory)
(Up to 3 years and $1,000,000)

Applications are sought to research, develop, and design state of the art I&C technologies for advanced control rooms and plant control and protection systems for advanced reactors in order to:

- Reduce I&C testing, validation and verification efforts associated with licensing requirements for common cause failure, design basis accidents, and cybersecurity, through methods that would support advance reactor safety features, such as passive safety.
- Rad-Harden electronics for digital based components. (e.g. PLC and FPGAs)
- Automate and enhance plant operation, such as remote operations or single control room workstation.

NEET-2.2: Advanced Online Monitoring and Diagnostics Technologies
(Federal POC – Suibel Schuppner & Technical POC – Craig primer)
(Eligible to Lead: University or National Laboratory)
(Up to 3 years and $1,000,000)

Applications are sought to develop and demonstrate advanced online monitoring for nuclear plant operation and maintenance to be integrated into new nuclear reactor designs. Applicants should:

- Demonstrate an optimal balance between cost and plant performance through a cost-benefit analysis for achieving reliability, availability, maintainability, and security.
- Integrate predictive analytics and risk informed condition monitoring, with business process applications, which would enable a transformational approach to supply chain and asset management.

NEET-2.3: Advanced Sensors and Communication
(Federal POC – Suibel Schuppner & Technical POC – Craig primer)
(Eligible to Lead: University or National Laboratory)
(Up to 3 years and $1,000,000)

Applications are sought to develop and demonstrate new sensors and instrumentation for advanced plant control, data analytics, and nuclear applications for advanced reactors. Applicants should:

- Develop advanced instrumentation and communication of data located in higher temperature, higher radiation reactor core environments found in advanced reactors.
- Develop smart multimodal measurement devices to measure unique and complementary parameters simultaneously.
- Develop new or unique application of materials for sensor development that support monitoring, controls, and communications within harsh nuclear reactor environments.
- Develop new radiation resistant sensors not currently under development for measurement of:
  - Local radiation and temperature (e.g. solid-state detectors, diamond thermistors)
  - Dimensional changes (specifically diameter and volume) and crack propagation,
  - Material properties, such as thermal conductivity, mechanical properties, thermal expansion, etc.)
  - Fission gas release (pressure and composition).
  - Other in-core parameters important to reactor safety and/or fuel performance.

PROGRAM SUPPORTING: NUCLEAR ENERGY ENABLING TECHNOLOGIES (NEET)

DIGITAL/ELECTRONIC NUCLEAR FIELD SUPPORT TECHNOLOGY (NEET-3)
(FEDERAL POC – SUIBEL SCHUPPNER & TECHNICAL POC – CRAIG PRIMER)
(ELIGIBLE TO LEAD: UNIVERSITY OR NATIONAL LABORATORY)
(UP TO 3 YEARS AND $500,000)

The Advanced Sensors and Instrumentation program seeks conceptual design applications for innovative technology for digital/electronic field support systems for nuclear facilities. These technologies should be

integrated and seamless able to enhance current state of the art technology used at nuclear facilities for real time measurements such as:

- Visual inspections and accountability
- Area radiation monitoring via remote monitoring or as part of personnel dosimetry
- Access and location monitoring – personnel access and security tracking
- Field worker "Head Up Display" to provide design/engineering information

The proposal should indicate how the proposed technology is or may be applicable to a nuclear facility. It should provide a plan for technology development and demonstration after the completion of this project.
Proposals should support the Department of Energy's (DOE) Office of Nuclear Energy's (NE) mission to advance U.S. nuclear power in order to meet the nation's energy needs by: 1) enhancing the long-term viability and competitiveness of the existing U.S. reactor fleet; 2) developing an advanced reactor pipeline, and, 3) implementing and maintaining the national strategic fuel cycle and supply chain infrastructure.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF-1)

NUCLEAR ENERGY-RELATED R&D SUPPORTED BY NUCLEAR SCIENCE USER FACILITIES CAPABILITIES (NSUF-1)

NOTE: FC-2.5 and **NEAMS-2** require NSUF access but can only be led by universities. Those workscopes can be found on page 77 and 81.

This workscope solicits applications for nuclear energy-related research projects focused on the topical areas described below. It is intended that these focused topical areas will change with each future CINR FOA. The focused topical areas are selected by NE's R&D programs (e.g. Nuclear Reactor Technologies, Fuel Cycle Technologies, and Nuclear Energy Enabling Technologies) with the explicit purpose to leverage the limited R&D funding available with access to NSUF capabilities. All applications submitted under this workscope will be projects coupling R&D funding with NSUF access. Projects requiring "NSUF access only" (see NSUF-2 below) or "R&D funding only" must be submitted under other appropriate workscopes. Applications submitted under this workscope must support the Department of Energy Office of Nuclear Energy mission. Capabilities available through the NSUF can be found on the website at nsuf.inl.gov.

The Office of Nuclear Energy (NE) supports the Department of Energy's HPC4 Materials (High Performance Computing for Materials) initiative to accelerate "industry discovery, design, and development of materials for severe environments by enabling access to computational capabilities and expertise in the DOE laboratories". NE's high-performance computing capabilities include Falcon at the Idaho National Laboratory. More information on computational resources can be found at NSUF.inl.gov. NE is seeking proposals for the development of innovative materials or material concepts for the extreme operating and accident environments expected in advanced reactor and fuel cycle technologies using the high-performance computing capabilities at the INL.

NSUF 1.1: TESTING OF ADVANCED MATERIALS OR ADVANCED SENSORS FOR NUCLEAR APPLICATIONS
(FEDERAL POC: SUIBEL SCHUPPNER & TECHNICAL POC: BRENDEN HEIDRICH)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 3 YEARS AND $500,000)

Proposals are sought for irradiation testing and post-irradiation examinations that support the development of advanced materials for sensors, and development of advanced sensors themselves to support NE's mission to enhance the long term viability and competitiveness of the existing fleet, to develop an advanced reactor pipeline, and to implement and maintain national strategic fuel cycle and supply chain infrastructure. **This funding does not support research and development activities to develop materials or sensors, but rather the irradiation of sensors and materials as described below.**

1) Advanced Materials for Sensors: Successful irradiation testing and post irradiation examination of candidate materials proposed for advanced sensors applications will include: a description of the materials; irradiation and post irradiation examination needs; the role of the materials in new sensors, controls, communications or associated applications.

2) Advanced Sensors: Successful irradiation and post irradiation examination of sensors and associated instrumentation will include: a description of the sensor and associated instrumentation and materials requiring irradiation and post irradiation examination; irradiation and post irradiation examination needs; and the purpose and application of the developed sensor in nuclear energy systems.

PROGRAM SUPPORTING: NUCLEAR SCIENCE USER FACILITIES (NSUF-1)

NSUF 1.2: IRRADIATION TESTING OF MATERIALS PRODUCED BY INNOVATIVE MANUFACTURING TECHNIQUES
(FEDERAL POC: TANSEL SELEKLER & TECHNICAL POC: BRUCE LANDREY)
(ELIGIBLE TO LEAD: UNIVERSITY, NATIONAL LABORATORY, OR INDUSTRY)
(UP TO 3 YEARS AND $500,000)

Products from advanced and innovative manufacturing and welding techniques can be proposed for evaluation of irradiation effects on material performance in support of NE's mission to enhance the long term viability and competitiveness of the existing fleet, to develop an advanced reactor pipeline, and to implement and maintain national strategic fuel cycle and supply chain infrastructure.

This funding does not support research and development activities to develop manufacturing and construction techniques, but rather evaluate the irradiation effects on material performance.

NUCLEAR SCIENCE USER FACILITIES (NSUF-2)

NUCLEAR SCIENCE USER FACILITIES ACCESS ONLY (NSUF-2)
(FEDERAL POC: TANSEL SELEKLER & TECHNICAL POC: RORY KENNEDY)
(ELIGIBLE TO LEAD: INDUSTRY ONLY)

Applicants interested in utilizing Nuclear Science User Facilities (NSUF) capabilities only should submit "access only" applications under this workscope. Applications must support the Department of Energy Office of Nuclear Energy's mission. Capabilities available through the NSUF can be found on the website at nsuf.inl.gov.

The Office of Nuclear Energy (NE) supports the Department of Energy's HPC4 Materials (High Performance Computing for Materials) initiative to accelerate "industry discovery, design, and development of materials for severe environments by enabling access to computational capabilities and expertise in the DOE laboratories". NE's high-performance computing capabilities include Falcon at Idaho National Laboratory. More information on computational resources available through NSUF can be found at https://nsuf.inl.gov/. NE is seeking proposals for the development of innovative materials or material concepts for the extreme operating environments expected in advanced reactor and fuel cycle technologies using the high-performance computing capabilities at the INL.
Experiments with synchrotron radiation may be proposed in applicable workscopes below. NSUF has access to the 6% of the available beam time at the X-ray Powder Diffraction beamline at NSLS-II.

NSUF-2.1: CORE AND STRUCTURAL MATERIALS

This element is primarily focused on fundamental understanding of irradiation effects in core and structural materials such as material aging and degradation mechanisms (e.g. fatigue, embrittlement, void swelling, fracture toughness, IASCC processes and mitigation), as well as developing alternate and/or radiation resistant materials for application in current and future fission reactors, and materials from alternate or advanced manufacturing techniques (including welding and joining). Proposed projects may involve R&D in the areas of material irradiation performance and combined effects of irradiation and environment on materials. Projects whose relevancy is based solely or primarily on fusion energy needs will not be considered. Proposals coupling experimental methods with modeling and simulation are highly encouraged.

NSUF-2.2: NUCLEAR FUEL BEHAVIOR AND ADVANCED NUCLEAR FUEL DEVELOPMENT

This program element is primarily focused on increasing our fundamental understanding of the behavior of nuclear fuels (including cladding) in reactor and research and development activities for advanced nuclear fuels and improving the performance of current fuels. Areas of interest include physics and chemistry of nuclear fuels,

NUCLEAR SCIENCE USER FACILITIES (NSUF-2)

irradiation and thermal effects on microstructure development and the effects on, for example, thermophysical and thermomechanical properties as well as chemical interactions. Advanced fuels applicability extends to fast spectrum transmutation systems, coated particle fuels for high-temperature reactor systems, and robust fuels for light water reactors including accident tolerant fuels. Activities should be aimed at irradiation experiments and post irradiation examination that investigate fundamental aspects of fuel performance such as radiation damage, amorphization, fuel restructuring, species diffusion and migration, and fission product behavior. Separate effects testing focused on specific V&V issues are encouraged. Proposals coupling experimental methods with modeling and simulation are highly encouraged.

NSUF-2.3: ADVANCED IN-REACTOR INSTRUMENTATION

This program element includes irradiation to support qualification of advanced in-reactor instrumentation for characterization of materials under irradiation in test reactors and for on-line core monitoring in power reactors. Applications should address the deployment and qualification strategy of radiation resistant sensors.

Development of techniques that are non-intrusive with respect to irradiation specimens, and nontraditional methods such as optical fibers and ultrasonic techniques as well as other incorporated wireless transmission techniques are encouraged. Proposals that also support the GAIN initiative, such as those involving development of advanced instrumentation, sensors, and measurement techniques for use in advanced reactors including molten salt reactors, sodium cooled fast reactors, lead cooled fast reactors, or high temperature gas reactors are encouraged. For MSR with dissolved fuel, an important and challenging problem is the ability to measure local chemical composition in real time at critical locations.

NOTE: Access to NSUF capabilities will require agreement and final signature to the User Agreement (copy provided in Appendix D. The terms and conditions of the User Agreement are non-negotiable and failure to accept the terms and conditions of the User Agreement will terminate processing and review of the FC-2.5, NEAMS-2, NSUF-1, or NSUF-2 applications. In order to ensure compliance throughout the application review process, applicants must indicate during the pre-application and full application submission that the User Agreement has been read, understood, and the terms and conditions are accepted. Further, submission of an pre-application and a full application indicates the applicant will comply and agree to the terms and conditions of the User Agreement. Upon award of an NSUF supported project, the User Agreement must be signed before activities will begin on the project.

Appendix C: Workscopes for U.S. University-led Integrated Research Project (IRP) R&D

PROGRAM DIRECTED: NUCLEAR ENERGY

INTERNATIONAL CHALLENGE PROBLEM FOR NUCLEAR ENERGY (IRP-NE-1)
(FEDERAL POC –JANELLE EDDINS & TECHNICAL POC – BOB HILL)
(ELIGIBLE TO LEAD: UNIVERSITIES ONLY)
(UP TO 3 YEARS AND $3,000,000 TOTAL U.S. PROJECT COST)

Background: The United States and Japan participate in a bilateral collaboration for R&D, the Civil Nuclear Energy R&D Working Group (CNWG). Both countries would like to explore measures to strengthen the U.S. Japan collaboration through an initiative to facilitate the use of a nuclear facility sharing. The initiative would be to support university researchers in the nuclear field to pursue bilateral research activities and in the process would allow greater access to use of research facilities in each country. This IRP request solicits proposals that would address nuclear research needs and also include participation of Japanese researchers (primarily from universities) on the IRP team.

IRP Goal and Objectives:

The Goal of this IRP request for proposals is to solve a significant problem which presents a challenge to expanded use of nuclear energy worldwide. The main objective of the IRP is to support education, development and training in multiple technical disciplines associated with the use of nuclear energy. The IRP requires participation by both U.S. and Japanese researchers.

IRP Scope:

Applicants must address the numbered elements listed below to be considered responsive to this IRP request for proposals.

1) Define a problem which presents a challenge to expanded use of nuclear energy worldwide. Examples of such problems are provided below:

   - Materials in GEN-IV reactors share a common environment extreme of irradiation but experience a dramatically different coolant environment (liquid metal, molten salt, etc.). While there is a relatively good understanding of material behavior in a separate irradiation or corrosion condition in some coolants, there is limited knowledge of material performance in coupled extremes of irradiation and corrosion, which dictates the safety of nuclear power plants and the development of future nuclear reactors. A fundamental understanding of the synergistic irradiation and electrochemical processes on material degradation mechanisms in various reactor environments using advanced experimental and computational tools is needed to predict the continuously evolving material properties over the reactor lifetime and to develop material design strategies to enhance their performance in advanced reactors.

   - Fuel-related safety criteria will be affected by changes to the fuel design as Accident Tolerant Fuel (ATF) concepts are developed. A key criterion that will be affected is the LWR fuel burn-up limit relating to fuel/cladding system response to reactivity initiated accidents (RIAs) and to loss-of-coolant accidents (LOCAs). One aspect that governs burnup limits is the ballooning and tearing of cladding under a RIA. In order to prepare for and assess impacts of ATF fuel changes on such safety criteria, improved understanding is needed of phenomena governing LWR fuel burnup limits. Techniques to facilitate 'accelerated or simulated aging' as an efficient assessment path for new metallic cladding, e.g. coated zircaloy and FeCrAl, concepts will also need to developed and validated.

PROGRAM DIRECTED: NUCLEAR ENERGY

- Improved accident tolerance for light water reactors or advanced reactors. Accident tolerance is intended to mean, elimination of the need for operator intervention and/or off-site emergency response for at least 72 hours without significant core damage. This work could include fuel cladding improvements, fuel design or modifications of the reactor internals.

- Construction and operating costs of a sodium-cooled fast reactor system could be reduced while maintaining or enhancing the safety of the power plant, e.g., by dramatically simplifying the design, improving performance, or by using advanced materials, such that the reactor system is at least cost competitive with currently available light water reactors, although it would be desirable to be even less expensive, on a per kW-hr basis. Significant experimental work to justify approaches such as less costly material selections and reduced margins in certain areas is needed to support development of these more cost competitive options. Considerable cost analysis work would be needed to show expected cost savings of a SFR design.

- Addressing important fuel cycle challenges including materials degradation, processing, safeguarding of nuclear facilities, and radioactive waste reduction will advance options for treatment of radioactive nuclear waste and reduction of waste activity and volume. Specific needs include the following:

  (1) Evaluation of radiation effect/degradation of the materials to be used in the fuel cycle facilities. The separation solvents such as ligands and molten salts are examples.

  (2) Development of remote monitoring system for nuclear materials and FPs in high temperature and high radiation field for enhancing the safeguard of nuclear facilities such as future fuel cycle facilities.

  (3) Development of innovative process to reduce radioactive waste by separation/removal of radionuclides such as actinides/fission products from spent fuel and waste materials.

The applicant can choose any or more than one of the above problems or define one on their own – however, the Applicant must elaborate on their understanding of the problem and how it meets the goal and objective of this request for proposals.

2) This IRP imposes International Collaboration Requirements with Japan which are in addition to those specified elsewhere in the Consolidated Innovative Nuclear Research Funding Opportunity Announcement (FOA).

   a. The project team must include at least two Japanese educational/research institutions. The team must include a lead Principal Investigator from a United States educational institution and personnel form a U.S. national laboratory. The scope of work for each collaborating institution must be clearly defined. The project team shall include a Co-Principal Investigator from Japan.
   b. All U.S. funds provided under this award must be used to support the efforts of the educational institutions and their non-university partners in the United States.
   c. Participation by Japanese researchers will be funded by the Government of Japan. Such funding should include at least one of the programs designated for funding by the Government of Japan. The point of contact for information regarding designated funding programs for Japanese researchers, Japanese facilities that could be used for potential projects, and Japanese University points of contact is Shoji Kasuga (kokusai-genshiryoku@mext.go.jp).

PROGRAM DIRECTED: NUCLEAR ENERGY
3) Facility Sharing: Nuclear Science User Facilities (NSUF) under the DOE Office of Nuclear Energy offers a wide range of material irradiation (reactors and ion beams) and material characterization facilities, and nuclear materials and fuels library. These facilities and resources would be of great interest to Japanese researchers. In return, Japan can make available their material research facilities to US researchers and share expertise and resources, beneficial to both parties. Facility use may include R&D/material irradiation for development of next generation nuclear reactors designs, nuclear fuel and/or materials. 4) The Applicant must describe how any intellectual property generated under the project will be handled. 5) Proposals are required to include collaboration with a Japanese University. 6) Project Deliverables: The list of proposed project deliverables must include: a. A detailed project schedule. b. If a software is developed as part of the project, a verification and validation plan for each software. c. A final project report which includes (i) A statement of the problem and a description of the solution (ii) The extent to which the problem was solved and remaining or follow-on work needed (iii) A description of the extent to which the project was successful in supporting education, development and training in multiple technical disciplines associated with the use of nuclear energy and (iv) Lessons learned and suggestions that can be used by DOE-NE and Japan in future work.

PROGRAM DIRECTED: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

EXTEND NEAMS TOOLS TO SUPPORT THE DEVELOPMENT OF THE VERSATILE TEST REACTOR (VTR) EXPERIMENTAL PROGRAM (IRP-NEAMS-1)
(FEDERAL POC – DAVE HENDERSON & TECHNICAL POC – CHRIS STANEK)
(UP TO 3 YEARS AND $4,000,000)

Advanced (non-LWR) reactors aim for significant advances in sustainability, safety, reliability, economics and non-proliferation. DOE supports the development of reactor concepts with improved economics for both electricity production and industrial applications. Many of the advanced reactor concepts being considered have wide-ranging coolant choices such as liquid metals (sodium, lead, lead-bismuth eutectic), helium, and molten salts are based on operation under fast spectrum. To address this priority, the DOE's Versatile Test Reactor (VTR) project is investigating a fast-flux irradiation facility that supports the experimentation and testing of a wide variety of fuel forms, materials, and advanced sensors for advanced nuclear energy systems. The initial concept for the VTR is a 300 MWth pool-type, sodium-cooled, metal fuel fast reactor system containing several static testing locations, closed loops (for experiments with different coolants), and rabbit systems. The VTR instrumentation suites and analytic capabilities will potentially result in petabytes of data over the next few decades. This presents a rare opportunity for modeling and simulation to both support the design of experiments as well as perform high fidelity analysis of experiments by correlating experiment response with the environmental conditions. Establishing and executing a modern strategy to simulate experiments and analyze measurement data is central to maximizing the impact of the entire VTR experimental program.

The goal of this Integrated Research Project (IRP) is to develop a tightly coupled research platform that enables comprehensive understanding of the multi-scale and multi-physics performance of the VTR reactor experiments through intelligent feedback between simulations and experiments. The primary focus of this effort is to extend the applicability and utility of the NEAMS tools to provide support in the development of the VTR Experimental Program through a tightly coupled collaboration between the selected IRP team, NEAMS, and the VTR Experimental group. The confluence of advanced modeling and simulation methods, data science and analytics, machine learning, and the exponential growth of computation capabilities have proven a powerful platform for modern scientific research and engineering. The VTR project presents a unique opportunity to integrate a modern data technology platform into the base design of a new advanced nuclear testing system.

The large IRP team effort should support both of the following two specific tasks:

1) **1) Accelerate extension of NEAMS computational tools to simulation of metallic fuel behavior and performance relevant to VTR and similar advanced reactor concepts, to include both high-fidelity and fast-running capabilities as needed.** Fast reactor fuels are typically designed to reach higher burnup than LWR fuels to take advantage of higher initial fissile loading and the breed-and-burn characteristics. Furthermore, swelling of materials is generally considerably greater in a fast neutron spectrum. Therefore, during the operation of an advanced reactor concept, the fuel assembly and rods may be subject to fast neutron irradiation induced swelling, thermal and irradiation creep, mechanical stresses from internal fission gas pressure, fuel handling and loading, thermal stresses from temperature gradients, flow-induced vibration and fretting, and fuel-cladding chemical and mechanical interactions. The fuel models must account for the impact of these factors on the integrity of the fuel assemblies and rods to assure that fuel system dimensions remain within operational tolerances and functional capabilities are maintained.

   Another important consideration is the analysis of fuel performance during anticipated operational occurrences and the postulated accidents that introduce an imbalance between the heat production and removal. The fuel design should provide assurance that the fuel is not damaged during the anticipated operational occurrences, and potential fuel damage during postulated accidents is not so severe that the core coolability is maintained and fuel damage does not propagate.

   The limiting performance concern for the metallic fuel rods during accidents is creep rupture of the cladding, accelerated due to fuel-cladding chemical interaction (FCCI). The mechanisms that influence

PROGRAM DIRECTED: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

FCCI in metallic fuels are fuel constituent migration, cladding attack by lanthanide fission products, and inter-diffusion between the fuel alloy and cladding, all eventually contributing to the formation of a low-melting point interaction product at the fuel-cladding interface. Since these processes are temperature and burnup dependent, fuel models should be able to predict fuel failure probability or margin to fuel failure as a function of burnup, fission-gas-plenum pressure, fuel-cladding interface temperature, and duration of a reactor transient during which the fuel is exposed to temperatures above the normal range.

Another critical need for advanced fuel modeling and simulation is validation, as a much as possible using existing relevant data, such as metallic fuel irradiation experiments performed in EBR-II and FFTF, and transient fuel testing at TREAT. DOE’s ART Fast Reactor Methods, Modeling and Validation R&D Program has been supporting development of EBR-II and FFTF metal fuel irradiation and physics analysis databases to collect and collate data needed to aid qualification of metallic fuel for future advanced reactor applications and validation of state-of-the-art codes and advanced methods for design and analysis of fast reactors. The selected experiments cover the range of fuel performance information, including prototype fuel behavior and failure mode during operational scenarios, fabrication parameters, lead tests, high temperature swelling behavior, and fuel-cladding mechanical interaction to demonstrate the viability of advanced metallic fuel forms for the U.S. vendors considering or designing advanced reactors using metallic fuels. The TREAT Test Database also covers the key metallic fuel experiments for margin to failure assessments.

Therefore, this task must include the following two important focus areas.

- Mechanistic model and transient simulation developments that incorporate or enhance capabilities for key phenomena, including
  - Fuel and cladding temperature distribution (as calculated by the NEAMS neutronics and thermal-hydraulics analysis tools),
  - Fuel-to-cladding and cladding-to-coolant heat transfer,
  - Axial burnup distribution in the fuel,
  - Thermal conductivity of the fuel and cladding for fresh and as-irradiated fuel,
  - Thermal expansion of the fuel and cladding,
  - Fission gas production, transport, and release,
  - Production and transport of lanthanide fission products,
  - Solid and gaseous fission product swelling,
  - Fuel constituent redistribution,
  - Cladding strain due to internal fission gas pressure and fuel-cladding mechanical interactions, and
  - Thinning of the cladding due to fuel-cladding chemical interactions.
- Validation of new models by leveraging the available EBR-II, FFTF, and transient fuel testing data that are relevant to both the VTR project as well as industry end-users.

2) **Accelerate coupled Neutronics and Thermal-Hydraulic Analysis capabilities to support the VTR experimental program and related advanced reactor commercialization.** Various uncertainties are involved in the predictions of reactor design parameters, which include theoretical and experimental analysis uncertainties, instrumentation uncertainties, manufacturing tolerances, correlation uncertainties, and method and simulation uncertainties. In order to assure integrity of fuel elements and reactor structures, and ultimately to protect the health and safety of the public and environment, the fuel, cladding, and coolant temperatures should not exceed the design limits with sufficient margins.

To that end, it is difficult to pin-point the exact eutectic formation temperature of a metallic fuel in an SFR because fuel-cladding eutectic formation is a complex phenomenon affected by fuel compositions, temperature, and irradiation histories. Thus, to account for theoretical and experimental uncertainties, the design limit of the eutectic formation temperature is lower than the value determined

PROGRAM DIRECTED: NUCLEAR ENERGY ADVANCED MODELING AND SIMULATION

theoretically or by empirical corrections. On the other hand, the peak cladding temperature increases from the nominal calculation to account for the stochastic errors of cladding thickness and compositions from manufacturing tolerances and uncertainties from modeling and simulations. After considering all uncertainties, the difference between the design limit and the peak cladding inner-wall temperature is defined as the minimum margin.

The gap from the nominal peak cladding inner-wall temperature to the eutectic formation temperature minus the minimum margin to limit denotes the uncertainties that are involved in the calculation of the design parameter, and in order to count the uncertainties in the reactor design, so-called hot channel factor methods have been developed and utilized since the early days of civil nuclear plant development. The hot-channel factor for a particular design parameter is defined by the ratio of the maximum value of that parameter to the nominal value. It is noted that the difference between the maximum value and nominal value is due to the uncertainties that are involved in calculating that parameter. If a factor is greater than unity for a particular parameter, the decimal part (i.e., HCF minus one) represents direct or stochastic uncertainties in the parameter.

In the United States, sets of hot-pin/channel factors have been developed for the Fast Flux Test Facility (FFTF), Clinch River Breeder Reactor (CRBR), and Experiment Breeder Reactor II (EBR-II), but additional attempts have not been made since these early SFR development programs were canceled in the mid-1990s. Development these factors with limited computing power and approximations in modeling and simulation methods resulted in the prediction of neutronics and thermal-hydraulic parameters with relatively large uncertainties, where some uncertainties could only be quantified by expensive experiments. The high-fidelity and multi-physics capabilities of the NEAMS program, accompanied by remarkable progress in the computing technologies during the last few decades, now make it possible to calculate most design parameters without support of mockups or experiments and reduce uncertainties significantly compared to the uncertainties involved in the earlier designs. Reduction of the uncertainties allows an increase in nominal parameters and safety margin, which in turn improves the economic competitiveness of the new designs.

Therefore, a key focus area will be application of NEAMS multiphysics analysis capabilities to the calculation of hot-pin and hot-channel factors, where the overall mature capability is expected to subsequently support the VTR experimental effort and help meet certain advanced reactor concept commercialization needs.

NEAMS codes, particularly those for fuels, neutronics, and thermal-hydraulics analyses, will be used in Tasks 1 and 2, as needed for the focus areas described in this scope. Should analyses be included in a proposal addressing aspects adjunct to the focus areas outlined in Tasks 1 and 2 (e.g., systems analysis), use of NEAMS codes would also be required. NEAMS code information may obtained from the Technical POC for this scope, as well as from the NEAMS website, https://neams.inl.gov/SitePages/Home.aspx. Proposals should include coordination by the IRP team with the NEAMS team and VTR project experimental groups.

PROGRAM DIRECTED: FUEL CYCLE

IRP-FC-1: USED NUCLEAR FUEL DISPOSITION: STORAGE & TRANSPORTATION (FEDERAL POC – JOHN ORCHARD & TECHNICAL POC – PETER SWIFT) (ELIGIBLE TO LEAD: UNIVERSITIES ONLY) (UP TO 3 YEARS AND $5,000,000)

Introduction

Dry storage of commercial spent nuclear fuel (SNF) is regulated by the U.S. Nuclear Regulatory Commission under Title 10 of the *Code of Federal Regulations* (10 CFR), Part 72 "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste." Some of the requirements under 10CFR72 are to prevent gross degradation of the cladding, maintain confinement of the radioactive material, and to maintain subcritical conditions, in part by maintaining the configuration of the spent fuel dry cask storage system (DCSS) internals (e.g., cladding, assembly hardware, and basket). An inert environment is desired to meet these requirements. This inert environment is obtained by removing the water from the cask and drying the SNF using either a vacuum drying or forced gas dehydration process (see for example ASTM C1553-16 "Standard Guide for Drying Behavior of Spent Nuclear fuel"). It is important to understand and quantify how much residual water may remain in a DCSS following drying.

Background

NUREG–1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," states that an accepted method for vacuum drying of canisters is to evacuate to a pressure less than or equal to 4×10^{-4} MPa [3 torr], with demonstration that the canister will maintain a constant pressure for 30 minutes after being isolated from the pumping system. Often, industry will evacuate to pressures much less than 4×10^{-4} MPa [3 torr] and consider the drying process complete if the pressure remains below 4×10^{-4} MPa [3 torr] when isolated from the pumping system for 30 minutes. However, the pressure may be increasing due to both temperature increase and off-gassing or vaporization of remaining water. Many factors such as materials of construction (e.g., porous neutron poison materials), local and total decay heat load, temperature variations in the system, rate of evacuation including hold points (important to prevent icing), assembly design (grid spacers, dashpots, instrument tubes, water tubes, etc.) and the presence of damaged fuel rods may affect the effectiveness of the drying process.

Objective

This IRP is meant to build upon and expand on, not duplicate, the work performed by a recently completed IRP at the University of South Carolina. The objective of this IRP is to quantify the amount of unbound liquid water and the amount of water vapor that remains in a canister following drying performed according to typical industry practices, both vacuum drying and forced gas dehydration. In particular, the sensitivities of the effectiveness of drying on the various factors listed above, with a major emphasis on the role of large thermal gradients in the system, is to be determined.

Considering these challenges, the research needs of this IRP shall include all the following:

- **Fuel Assembly Mockup**

  To undertake these tests, a vacuum drying system and a forced-gas dehydration system similar to those used in the industry should be acquired or built, then employed on specialized canister and fuel assembly mockups. The fuel assembly mockups shall physically represent locations where water could be difficult to remove from prototypic assembly designs (e.g., pressurized water reactor 17x17 and boiling water reactor 10x10 assemblies). These should include breached rod(s) with the size and location of the holes based on operational experience for damaged fuel. Other locations shall include the dashpot region of the guide thimble tubes for pressurized water reactor assemblies, water rods for boiling water reactor assemblies, and creviced regions associated with assembly hardware such as grid spacers, nozzles, and tie plates. The guide thimble tube shall also include a mock-up rod (representing a burnable poison rod, control rod or poison rod)

PROGRAM DIRECTED: FUEL CYCLE

inserted in the guide tube to partially block the guide tube.

As needed, multiple mockups may be employed to represent different assembly designs or the range of features from different designs may be incorporated in a single mockup. The mockup shall be kept at full-length and shall have the ability to be heated to represent the decay heat load of SNF. The total decay heat shall be capable of being varied to represent both aged (long-cooled) and young (two years cooled) SNF. The thermal profile of the mockup shall be variable to simulate those found in actual SNF DCSSs.

- **Canister Mockups**

Full-sized canister mockups are not required for the test program, but they shall be able to accommodate full-length mockup assemblies. Canister mockups could be fabricated from pipe segments or other cylindrical structures fitted with bolt-on lids to allow for insertion and removal of the mockup assembly. Except for any modifications that are needed for making measurements, the ports for connection between the canister and drying system, as well as the configuration of the vacuum siphon tube, should be similar to those in industry systems. Heaters, internal or external, or insulation may be used to develop both axial and radial temperature gradients similar to those found in SNF DCSSs. Accurate measurements of the temperature profiles in the system are required. Similarly, the canister system pressure must be recorded from the start of the drying process, through the pressure rebound test, after He backfill, and until a thermal equilibrium is achieved.

- **System Testing**

The tests will involve the performance of drying operations in a manner consistent with industry practice, after which the quantity of water, both as liquid and as vapor, remaining in the canister will be determined. In a series of drying runs, specific variations of certain parameters should be made to quantify how these affect the quantity of residual water. Methods for quantifying the remaining water shall be validated and multiple methods are preferred to determine the sensitivity and accuracy of each method.

Vacuum drying is typically performed in a stepwise approach to progress down in pressure, thereby reducing the likelihood of ice formation by limiting the pumping speed and providing time for the system to equilibrate. Within the industry procedures, however, there are differences in specifications such as the number of hold points and the end pressure. Therefore, the drying runs must include variations in these parameters to envelop the range of industry standard practices. Forced-gas dehydration tests should include a range of inert gases (He, N, etc.) consistent with those used in practice. Variability in gas temperature and pressure shall be included in the testing matrix for the forced-gas dehydration tests.

While individual, separate effects tests may be run, drying sensitivity studies shall be performed on a system that has been fully flooded for at least 48 hours prior to performing the drying operation.

TASKS TO BE PERFORMED

- **Task 1:** Development of Test Plan
- **Task 2:** Development of Analytical Models for Drying Simulation
- **Task 3:** Setup and Verification of Test System
- **Task 4:** Performance of Drying Tests
- **Task 5:** Complete Project Report

DELIVERABLES

Specific deliverables shall include:

- **Approval of Test Plan by an Independent, External Review Committee that contains DOE and industry representatives**
- **Quarterly Progress Reports**
- **External Review Committee Review every 6 months**
- **Final Project Report**